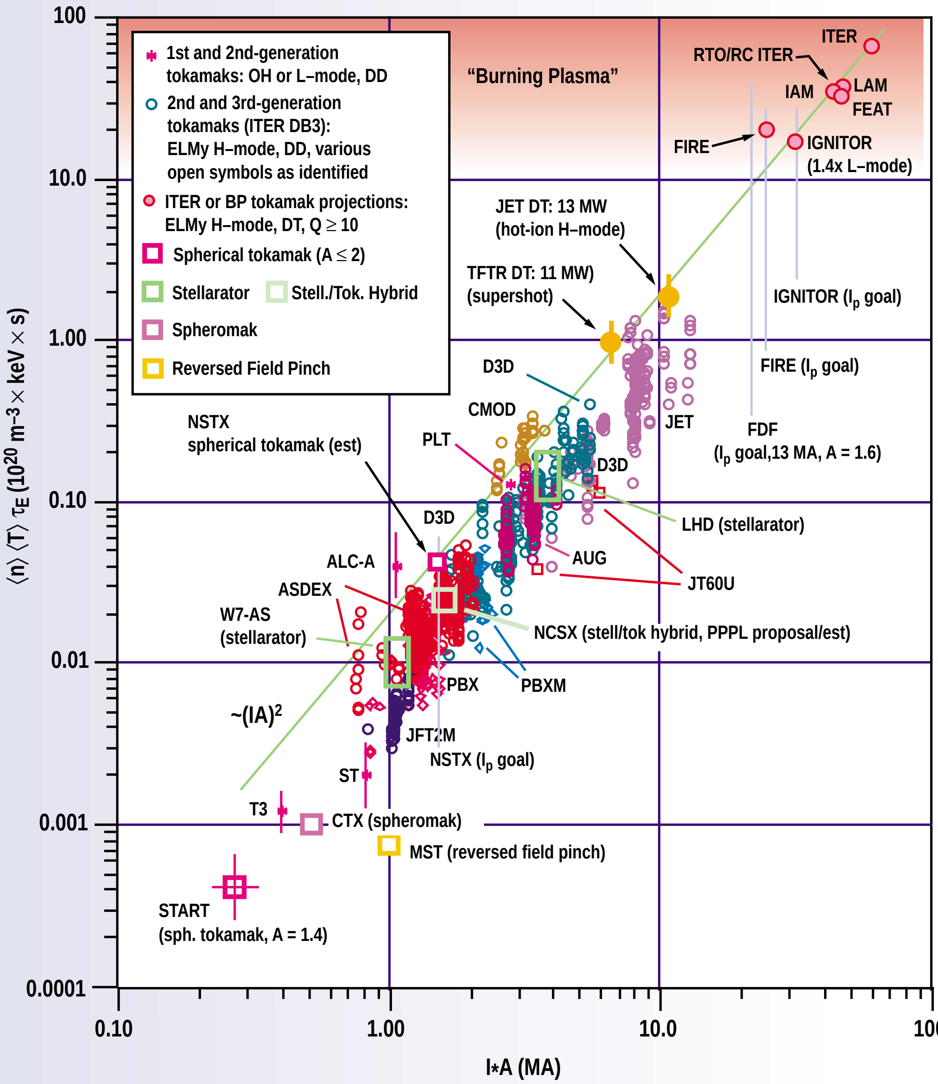
FUSION RESEARCH AT GENERAL ATOMICS



ANNUAL REPORT

OCTOBER 1, 1998 THROUGH SEPTEMBER 30, 1999

ADVANCED FUSION TECHNOLOGY RESEARCH AND DEVELOPMENT

GENERAL ATOMICS

GA–A23325
UC–420

ADVANCED FUSION TECHNOLOGY RESEARCH AND DEVELOPMENT

ANNUAL REPORT TO THE U.S. DEPARTMENT OF ENERGY

OCTOBER 1, 1998 THROUGH SEPTEMBER 30, 1999

by
PROJECT STAFF

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Cover Photo: Actual and predicted fusion performance ($nT\tau_E$, volume-average temperature and density basis) for various tokamaks and other toroidal confinement experiments. The parameter IA (plasma current × aspect ratio) is found to be a relatively good indicator of fusion performance (see Section 5).

CONTENTS

LIST OF FIGURES

Section 1

ADVANCED FUSION TECHNOLOGY RESEARCH AND DEVELOPMENT OVERVIEW

1. ADVANCED FUSION TECHNOLOGY RESEARCH AND DEVELOPMENT OVERVIEW

The General Atomics (GA) Advanced Fusion Technology program seeks to advance the knowledge base needed for next-generation fusion experiments, and ultimately for an economical and environmentally attractive fusion energy source. To achieve this objective, we carry out fusion systems design studies to evaluate the technologies and materials needed for next-step experiments and power plants, and we conduct research to develop basic and applied knowledge about these materials and technologies. GA's Advanced Fusion Technology program derives from, and draws on, the physics and engineering expertise built up by many years of experience in designing, building, and operating plasma physics experiments. Our technology development activities take full advantage of the current DIII–D program and facility.

The following sections summarize GA's FY99 work done in the areas of Fusion Power Plant Design Studies (Section 2), Advanced Liquid Plasma Facing Surfaces (Section 3), Advanced Power Extraction Study (Section 4), Next Step Fusion Design (Section 5), Plasma Interactive Materials (Section 6), Radiation Testing of Magnetic Coil (Section 7), Vanadium Component Demo (Section 8), RF Technology (Section 9) and Inertial Fusion Energy Target Supply System (Section 10). Our work in these areas continues to address many of the issues that must be resolved for the successful construction and operation of next-generation experiments and, ultimately, the development of safe, reliable, economic fusion power plants.

The work was supported by the Office of Fusion Energy Sciences, Facilities and Enabling Technologies Division, of the U.S. Department of Energy.

Section 2

FUSION POWER PLANT DESIGN STUDIES

2. FUSION POWER PLANT DESIGN STUDIES

Significant progress in physics understanding of the reversed shear advanced tokamak regime has been made since the last ARIES-RS study was completed in 1996. The 1999 work aimed at updating the physics design of ARIES-RS, which has been renamed ARIES-AT, using the improved understanding achieved in the last few years. The new work focused on:

— Improvement of beta-limit stability calculations to include important non-ideal effects such as resistive wall modes and neo-classical tearing modes
— Use of physics based transport model for internal transport barrier (ITB) formation and sustainment
— Comparison of current drive and rotational flow drive using fast wave, electron cyclotron wave and neutral particle beam
— Improvement in heat and particle control
— Integrated modeling of the optimized scenario with self-consistent current and transport profiles to study the robustness of the bootstrap alignment, ITB sustainment, and stable path to high beta and high bootstrap fraction operation.

We have reaffirmed some previous conclusions and discovered new insights to key issues that are expected to have impact on the ARIES-AT design. In summary:

In stability optimization

— Ideal low n modes are stabilized by a conducting wall at ~ 1.2 a
— Beta is limited by high n ideal ballooning modes near the plasma outer region
— Rotational drive and radially localized off-axis current drive are essential for stabilization against resistive wall modes and neo-classical tearing modes

In transport and current drive modeling

— The initial 13.2 MA, β_N=5.6 design produces too much alpha heating
— Physics-based modeling with ITB indicates a smaller device should be considered

In divertor heat exhaust

— High radiated fraction of the total exhausted power (>0.5) is essential to keep the peaked inboard and outboard heat fluxes at a manageable level (< 10 MW/m^2)
— It is essential to accurately maintain the double-null magnetic balance (to ~0.5 cm)

Details are available in GA report GA-C23336.
Based on these findings, we recommend the following future work:

- Explore dependence of pressure and bootstrap profiles on density profile and rotation profile
- Look for self-consistent pressure, bootstrap, density, and rotation profiles giving near optimal stability
- Calculate rotation needed for stabilization of resistive wall mode and calculate the resulting neutral beam requirements
- Perform a power balance
- Calculate UEDGE 2D solution for edge profiles
- Determine stabilization requirements for neoclassical tearing modes

PUBLICATIONS/REPORTS

1. V.S. Chan, *et al*., “Physics Optimization of the ARIES-RS Fusion Power Plant,” Presented at the 41st Annual Meeting of the Division of Plasma Physics (APS), Seattle, Washington, 1999, Paper CPI 94.
2. V.S. Chan, *et al*., “Advanced Fusion Power Plant Studies Annual Report for 1999,” General Atomics Report GA-C23336, January 2000.

Section 3

ADVANCED LIQUID PLASMA-FACING SURFACES (ALPS)

3. ADVANCED LIQUID PLASMA-FACING SURFACES (ALPS)

MODELING

A study was initiated utilizing the GA reactor systems code to assess the impact of a liquid divertor and first wall in an ITER-type fusion reactor. The approach developed was to couple 1-D scrape-off-layer density and temperature profiles, from the LLNL's UEDGE code, to GA's ONETWO core transport code. The goal was to determine the impact of changes in density and temperature profiles on the cost of electricity.

The preliminary studies were done with the ITER geometry due to the availability of ITER-relevant transport data from DIII–D. UEDGE runs were completed for a low recycling ITER case (to simulate liquid lithium divertors) and a standard recycling case. A program was written to reduce the grid resolution and to reformat the data in the equilibrium file. Initially, the transport coefficients used in ONETWO were based on benchmarked data from DIII–D runs done with shapes similar to those specified for ITER. Transport coefficients had to be adjusted locally in order to get a stable solution. The ONETWO solutions for the low and standard recycling cases were not significantly different because the density in the low recycling UEDGE runs had been held to approximately the same value as in the standard case by introducing a particle source in the edge.

Based on this study, we now know that we can produce a set of quasi-stable, self-consistent, radial profiles for the systems code using ONETWO. We also learned that the central profiles are more sensitive to the shape of transport coefficient profiles in the pedestal region than they are to the boundary conditions on temperature and density. For these studies, the radial profiles of the transport coefficient had to be significantly modified from the originally benchmarked DIII–D profiles in order to get quasi-stable solutions in ONETWO. This implies that to approximate a simple scaling of the transport in ITER for these studies, similar DIII–D plasma scaling is inappropriate.

Since the scaling approach does not work, a more appropriate method may be to start with a theory-based model. Several theoretical transport models are available in ONETWO, including the Dorland-Kotchenruther IFS model, but none have pedestal-specific physics incorporated yet, so the accuracy of this approach is unclear. However, we will continue the evaluation by using a simple density scaling study with ONETWO. We will then decide whether to proceed with a more comprehensive development of ONETWO, with complete input from UEDGE, after analyzing results from the simple density scaling study.

REACTOR SYSTEM STUDY

The GA reactor system code was used to project high power density ARIES-RS- and ARIES-ST-type reactor designs. Results showed that at a power output of 2 GWe and a neutron wall loading of 7 MW/m^2, both designs can have a cost of electricity at <65 mill/kWh, which is comparable to the projected cost of coal with the inclusion of CO_2 tax or cost of CO_2 sequestration. Four tokamak designs, combining normal and superconducting coils, with and without Kr addition for radiation, were completed and distributed to the ALPS and APEX teams. This GA-system code is now set up to evaluate other design options of the ALPS and APEX programs.

ATOMIC DATA COORDINATION

GA joined the ADAS atomic database consortium in December, 1998. The ADAS-IDL software was installed on a local workstation which can be accessed by members of the ALPS edge-modeling group.

PISCES MODELING

A 2-D version of GA's MCI kinetic SOL modeling code was developed for modeling CH_4 gas injection experiments in PISCES. The goal is to use the PISCES data to validate the methane ionization/dissociation and transport models used in MCI. Initial results, using a Stangeby-type velocity diffusion model for parallel molecular transport, showed a e^{-1} CH penetration depth which was approximately twice the measured penetration in PISCES. This discrepancy may be due to the use of the reaction rates supplied by the original Ehrhardt and Langer model. We will continue to investigate the discrepancy between experimental data and modeling results.

SUMMER STUDENT

In collaboration with the SDSU Student Foundation, a third-year undergraduate physics student was hired during the summer to assist with MCI programming tasks.

CONFERENCES/MEETINGS

A talk on the modeling of chemical sputtering in DIII–D with MCI was given at the October 1998 U.S.-Japan High Heat Flux Components Workshop in Japan.

Section 4

ADVANCED POWER EXTRACTION STUDY (APEX)

4. ADVANCED POWER EXTRACTION STUDY (APEX)

REFRACTORY ALLOY HELIUM BLANKET DESIGN

In FY99, the preliminary design of a high performance He-cooled W-5Re alloy FW/blanket was completed. A separate first wall that is permitted to flex under heating and a lithium pool configuration was selected, as shown in Fig. 4-1. Due to the lack of irradiation data, conservative assumptions for selecting the tungsten alloy properties were used. The compatibility of tungsten alloys with oxygen as the primary helium impurity was considered. Commercially available solid gettering modules can maintain the impurity level to << 1 appm and prevent embrittlement. Based on the results of the selected analyses done, the FW/blanket design could meet the material temperature and structural design limits, provided that the peak structural loading during disruptions can be mitigated. The 1-D tritium-breeding ratio goal of 1.43 can be reached with a Li-6 enrichment of 35%, but the presence of induced radioactivity will not allow the tungsten alloy components to meet the criteria for classification as low-level waste. The tungsten alloy will generate a high level of afterheat, but with the tritium extraction system operating, long-term accident temperatures can be maintained below 800°C. A cold trap process with protium added to the lithium could be used for tritium extraction. At a combined cycle gas turbine (CCGT) gross thermal efficiency of 57.5%, a superconducting reactor with an aspect ratio of 4 and an output power of 2 GWe is projected to have a cost of electricity (COE) of 55 mill/kWh.

Critical issues were identified. Continuing evaluation of some of the critical issues during the next phase of the APEX design study is planned.

The notion of utilizing the CDX-U device at PPPL to investigate the interaction of liquid lithium and plasma in a toroidal device was conceived and suggested to DOE.

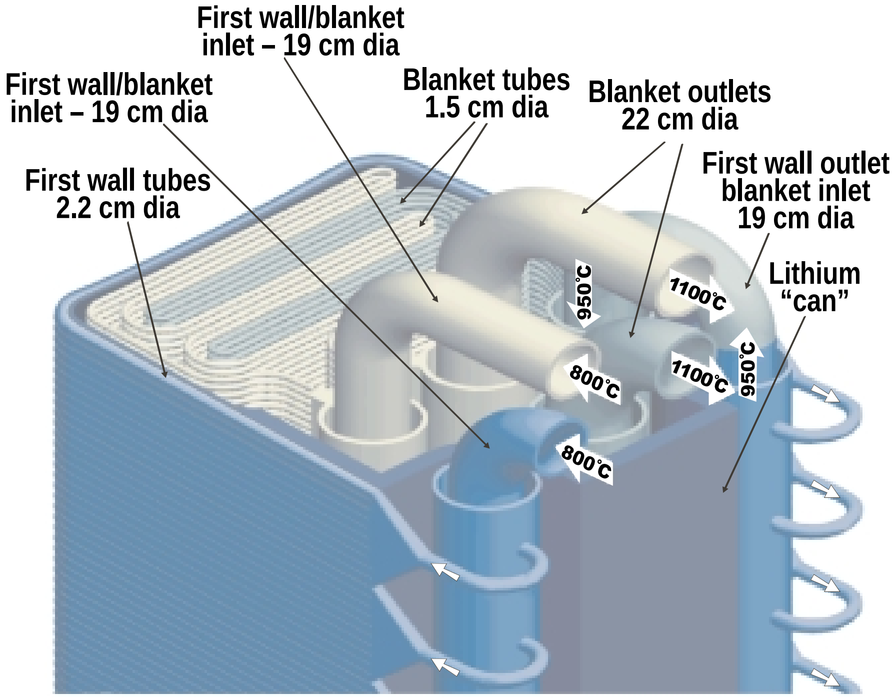


Fig. 4-1. APEX helium-cooled blanket concept. One of three "cans" per outborad sector shown with top removed.

CONFERENCES/MEETINGS

1. GA was represented at the various APEX project meetings throughout FY99.
2. 5th Japan/China Workshop on Materials for Advanced Energy Systems and Fission and Fusion Engineering, Xian, China, November 1998. C. Wong presented summaries of the ALPS and APEX programs.
3. ISFNT-5 meeting, Rome, Italy, September 1999.
4. 4th U.S./Japan Fusion High Power Density Workshop, Santa Fe, New Mexico, November 1999.

PUBLICATIONS/REPORTS

1. C. Wong was appointed editor of the 1999 APEX interim report. The report was finished on schedule. GA wrote the chapter on the refractory alloy helium-cooled blanket design.
2. "Helium-cooled Refractory Alloys First Wall and Blanket Evaluation," presented at the September 1999 ISFNT-5 meeting in Rome, Italy.

Section 5

NEXT STEP FUSION DESIGN

5. NEXT STEP FUSION DESIGN

This task provided physics analysis and other scientific and technical input to Next Step Options (NSOs) Studies for the U.S. Fusion Science Program. Emphasis in this work is on options (design candidates) to obtain plasma behavior at high energy gain and for long duration operation pulses. There were two subtasks included in the scope as originally set in late FY99: the first to provide definition of physics and plasma operation objectives, physics and plasma science assessments and definition of physics and other design requirements for U.S. NSO studies; the second to observe and report the progress of the ITER Reduced-Technical-Objectives/Reduced-Cost (ITER RTO/RC) design activities.

These activities were projected to comprise approximately 0.25 FTE each and were to be conducted on an approximately constant level-of-effort basis for each subtask. John Wesley was the principal investigator at GA. In accordance with OFES guidance received during the 3rd Quarter of FY99, participation in ITER Physics Expert Group meetings and ITER Joint Central Team organized activities were suspended and the effort redirected into the first (U.S. NSO studies) subtask.

All subtasks within the original scope have been completed as planned or as later amended to conform to OFES guidance. A summary of activities conducted during FY99 and key findings follow.

ITER REDUCED-TECHNICAL-OBJECTIVES/REDUCED-COST DESIGN OBSERVATION

John Wesley worked in collaboration with Paul Rutherford (PPPL) in preparing a set of generic physics and fusion technologies issues and corresponding metrics that would potentially apply to candidate NSOs, both U.S. and international. This work was subsequently presented at the U.S. NSO meeting at PPPL (see below) and incorporated into an invited presentation at the 1999 Fusion Summer Study (Snowmass meeting) entitled “Next Step Options: Physics Issues and Performance Measures.”

Wesley attended the RTO/RC ITER Point Design Meeting and reported the results at the U.S. NSO Workshop held at PPPL.

Work on this task during FY99 showed that there are several common key physics issues for proposed U.S. NSOs and for RTO/RC ITER, including questions about the basis for the projection of energy confinement to more collisionless reactor-regime plasmas and about the performance-limiting effects of neoclassical tearing modes (NCTMs) in this regime. Other common issues and concerns include the likely need to actively control/suppress the growth of NCTMs with spatially-localized electron cyclotron current drive (ECCD) and questions about the need to provide similar active control/stabilization of resistive wall modes (RWMs) that develop in advanced tokamak (AT) operation modes wherein the plasma beta exceeds the so-called “wall-at-infinity” ideal MHD stability limit.

Work on this task was stopped during the 3rd Quarter of FY99 per OFES direction.

DEFINITION OF PHYSICS AND PLASMA OPERATION OBJECTIVES FOR U.S. NSO STUDIES

This task focused on quantifying physics and plasma science assessments and definition of physics and other design requirements for U.S. NSO studies, especially as embodied in the FIRE (Fusion Ignition Research Experiment) tokamak burning plasma experiment concept being developed by the FIRE national design team centered at PPPL. The FIRE concept employs high-field (10 T) resistive copper magnets to achieve high-Q ($Q = P_{fusion}/P_{auxiliary} \cong 10$) operation in a DT plasma. Such operation, commonly termed “burning plasma” operation, is intended to explore the physics and technology aspects of largely self-heated DT plasmas capable of producing appreciable levels of sustained fusion power. In the case of FIRE, thermal limitations on magnet pulse length and on the ability of inertially-cooled plasma-facing-components to absorb fusion energy act to limit DT burn pulse durations to about 10 s. However, as the physics and plasma performance assessment studies done in this task show, this pulse length is sufficient to address many of the fundamental physics issues of burning high-Q plasmas, including the effects of energetic alpha-particle populations and the dynamic effects of self heating.

FIRE and ITER operate in similar reactor-scale physics regimes and share many common physics and plasma operation attributes, including a common need to obtain and sustain H–mode operation, wherein a an energy transport barrier formed at the plasma edge provides the plasma energy confinement increment needed to obtain Q = 10 in a FIRE or ITER plasma. Figure 5-1 shows how H–mode access and sustainment requirements for FIRE constrain the accessible plasma operation regions in plasma density-temperature space. As the figure shows, to obtain H–mode before appreciable fusion power is produced, FIRE plasma operation must first commence at low density ($\cong 10^{20}$ m^{-3}). Once H–mode is obtained, increase of plasma density to a final value of 4×10^{20} m^{-3} results in a final fusion power of about 200 MW. This dynamic H–mode access scenario strategy is essentially identical (except for the magnitude of the density)

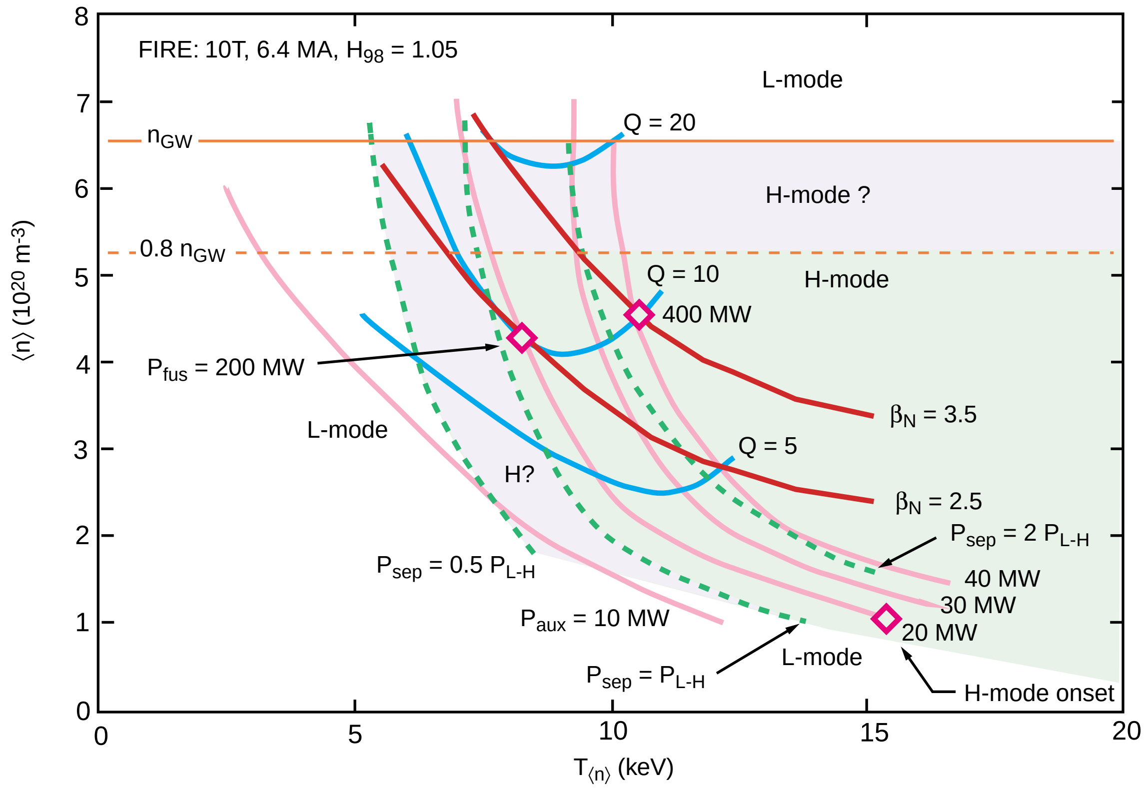


Fig. 5-1. FIRE plasma density/temperature plot with H–mode and L–mode domains indicated. The final FIRE operation point at 4×10^{20} m^{-3} density and average temperature of 8.5 keV lies near the lower density bound of the sustainable H–mode domain.

to that needed for ITER RTO/RC, and sustainment of H–mode in a Q = 10 operation regime is found to be a physics issue of similar concern for both experiments.

Task activities included continuing participation in the FIRE national design study effort, and participation in the Burning Plasma Physics subgroup of the MFE Study Group at the 1999 Fusion Summer Study (Snowmass Workshop). The most notable of the Group's findings was the conclusion (supported by a large majority of the participants) that the tokamak concept is technically ready to undertake a burning plasma experiment. The tokamak was the only MFE concept judged ready to receive such endorsement.

Figure 5-2 shows data presented at Snowmass to show the proximity of various toroidal magnetic fusion concepts to the burning plasma regime, indicated by the shaded region. The threshold of this domain is at $\langle n\rangle\langle T\rangle\tau_E \cong 1.5 \times 10^{21}$ m^{-3} keV s. Here the $\langle n\rangle$ is the volume-average plasma density, $\langle T\rangle$ is the volume-average plasma density and τ_E is the the plasma energy confinement time. A commerical toroidal magnetic fusion reactor typically projected to require $\langle n\rangle\langle T\rangle\tau_E \cong 10^{22}$ m^{-3} keV s.

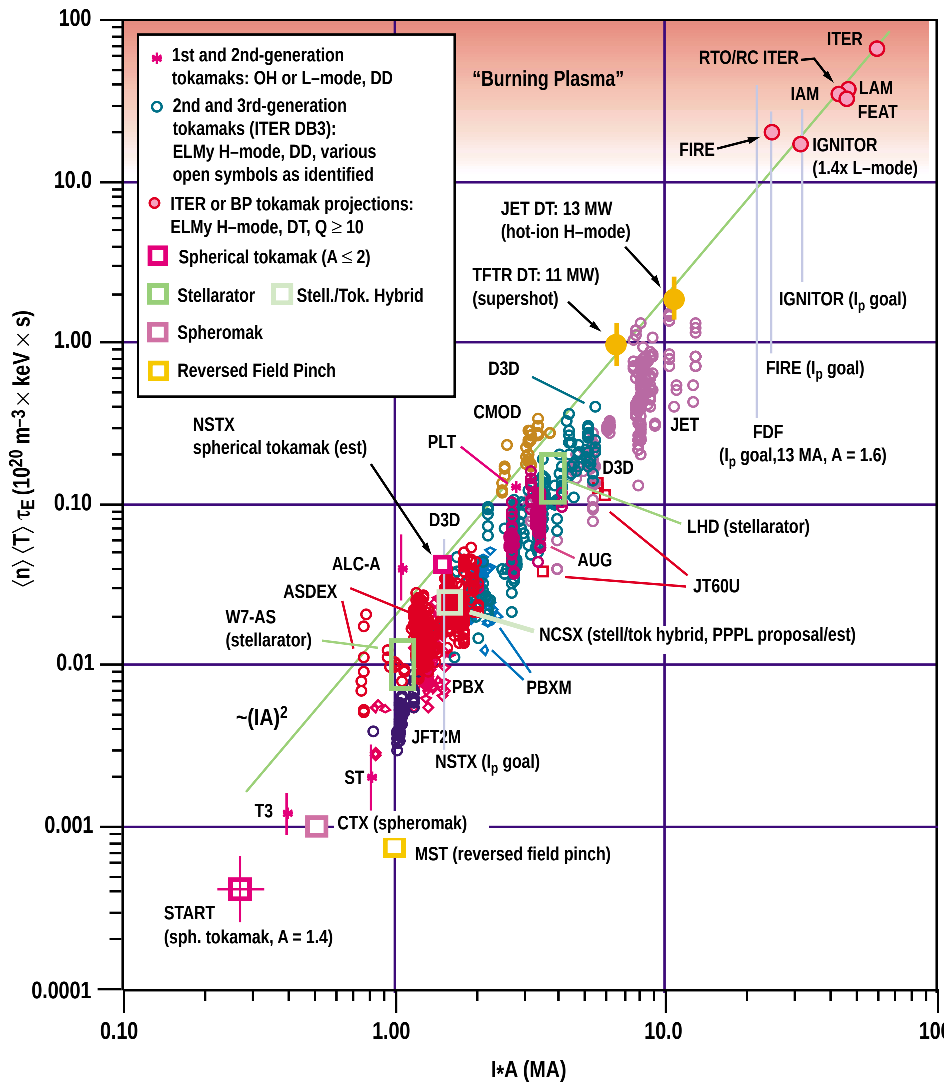


Fig. 5-2. Actual and predicted fusion performance ($nT\tau_E$, volume-average temperature and density basis) for various tokamaks and other toroidal confinement experiments. The parameter IA (plasma current × aspect ratio) is found to be a relatively good indicator of fusion performance

The Group debated the extent to which burning plasma studies conducted in a tokamak would be applicable to other toroidal and alternate MFE concepts that are not yet ready for a burning plasma class of experiment. The Group's finding was that results of burning plasma studies in a tokamak would, if quantified in terms of fundumental science understanding, be applicable to other toroidal MFE concepts, subject to the caveat that such application in another concept requires prior understanding of how the fundemental aspects of fusion science (e.g., MHD stability, turbulent transport, etc.) affected by a burning plasma apply to the concept in question. Given such understanding, the Group concluded that results from a tokamak burning plasma experiment could yield understanding applicable to a wide range of toroidal MFE concepts. One germ of this conclusion can be seen in Fig. 2, where nTτ in tokamaks, spherical tori and stellarators are all found to be reasonably well described by a common IA empirical scaling.

MEETINGS/CONFERENCES

1. RTO/RC ITER Point Design Meeting, San Diego, California, January 1999.
2. U.S. Next Step Options Workshop, Princeton, New Jersey, March 1999.
3. 1999 Fusion Snowmass Study (Snowmass Workshop), Snowmass, Colorado, July 1999.

PUBLICATIONS/REPORTS

P. Rutherford and J. Wesley, "Next Step Options: Physics Issues and Performance Measures," June 1999.

Section 6

PLASMA INTERACTIVE MATERIALS (DiMES)

6. PLASMA INTERACTIVE MATERIALS (DiMES)

Li EXPOSURE EXPERIMENT

The safety aspects of the Li-DiMES experiment were reviewed and authorization to proceed was obtained. The initial lithium exposure, a 0.25 mm thick sample, was done in June. The sample was prepared by SNL-CA, shipped to GA under argon atmosphere, and transferred to the DiMES sample changer using a glove bag and argon cover gas. The shiny metallic lithium surface was maintained during the transfer.

Three different plasma environments were recorded in a piggyback experiment:

1. The sample was exposed to the outer strike point during four ELMy H–mode plasma shots. Good data of neutral lithium and singly ionized lithium were collected.
2. The sample was exposed to private flux plasma. Bright Li I emission was recorded. While in the private flux zone, the plasma was typically in a high density divertor operating mode with low peak heat flux. The Li I emission may be a result of charge exchange neutrals.
3. The sample was exposed to a high power deposition MHD event (locked mode). Li I and Li II lines were recorded at the edge as shown in Fig. 6-1, and Li III was observed in the plasma core. As shown in Fig. 6-1, there is evidence of melting and displacement of the lithium surface, possibly from J×B forces.

Detailed plasma and spectroscopic data were distributed to DiMES collaborators for analysis. Results from these exposures will provide guidance on the planning of dedicated experiments, and have provided support on modeling benchmarks for the ALPS program and on the study of impurity transport physics.

SOLID SURFACE EROSION AND REDEPOSITION EXPERIMENTS

A vanadium and tungsten coated sample was exposed to the inner strike point of a partially detached DIII–D divertor plasma. A multi-chord divertor spectrometer identified several lines (C I, C II, C III, CD molecular band, B I, and B II) during repeated discharges. It appears from the floor probes data that significant particle flux reached the sample, and that the density and temperature are similar to the outboard strike point detached plasma parameters. The CD molecular band was either very weak or not measurable. In conjunction with PISCES results from UCSD, suppression of chemical sputtering by boron from wall conditioning was confirmed.

The erosion and redeposition data related to the concentration of trapped deuterium were reviewed and similar erosion and redepostion patterns were identified for all major tokamaks as shown in Fig. 6-2. This further supports the severe operational limits projected for next-step DT tokamak burning plasma devices that employ graphite tiles, due to a persistent inboard and outboard erosion and redeposition asymmetry. At an average neutron wall loading of 1 MW/m^2, the projected outer divertor plate net erosion is 10–30 cm/burn-year, and the projected tritium co-deposition rate is ~ 1 kg/m^2/burn-year.

DIII–D CHEMICAL EROSION

An *in-situ* atomic and molecular spectroscopy database has been compiled on carbon chemical erosion over the last seven years of DIII–D operation. The database shows that the carbon chemical erosion yield has decreased a factor of ~20, supposedly due to the 2.5 μm of boronization layers applied over the same period. This is in agreement with tests on DIII–D tile sections in the PISCES device at UCSD. Overall, the total carbon source rate (physical + chemical sputtering) appears to have decreased about a factor of four since 1992. Conversely, no concomitant decrease in the core plasma contamination has occurred during the same period. Explanations for these observations are being explored.

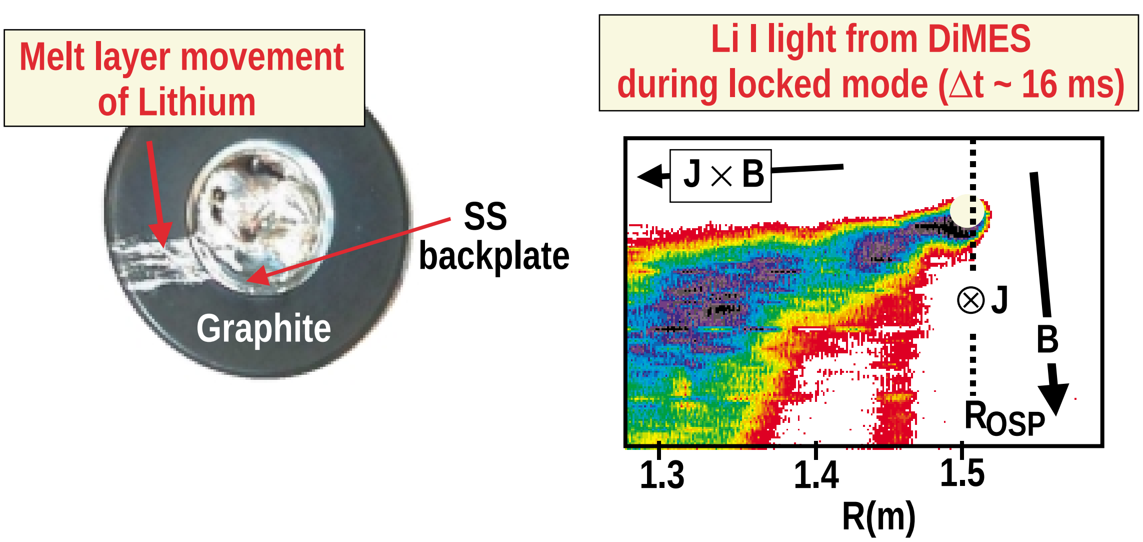


Fig. 6-1. First exposure of a Li sample in DIII–D.

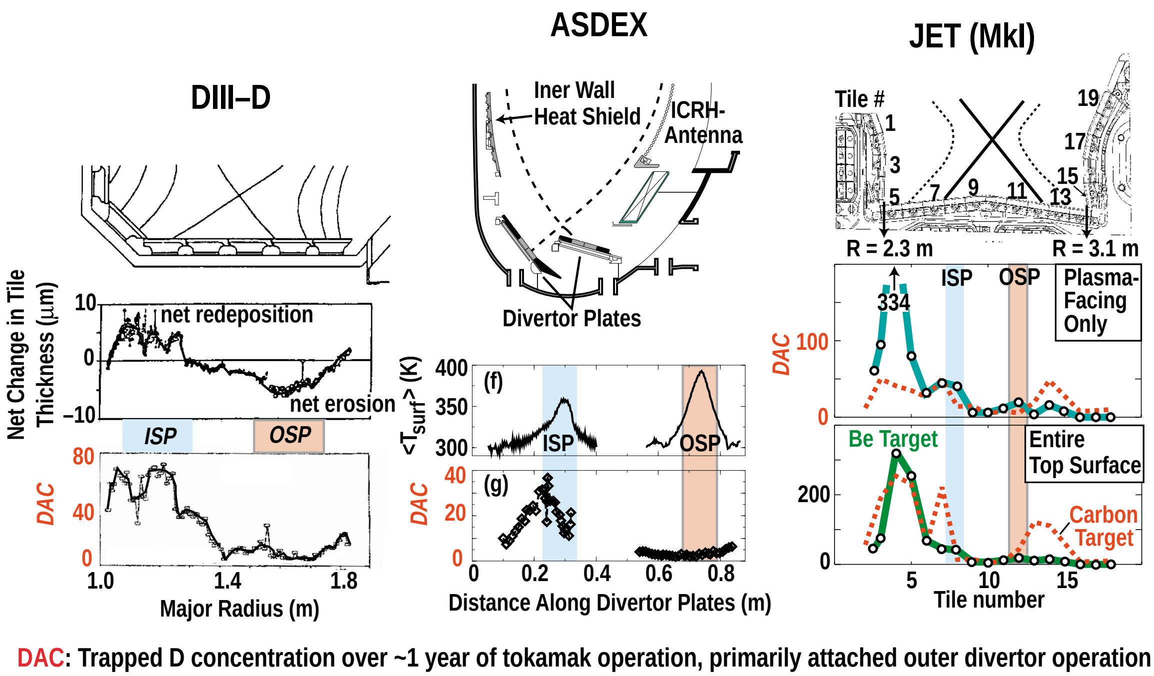


Fig. 6-2. Review of erosion/redeposition data related to trapped deuterium concentrations led to identification of similar erosion/redeposition patterns on all major tokamaks.

TUNGSTEN EXPOSURE

A tungsten sample was exposed to ~15 partially detached divertor plasma discharges during a plasma flow measurement experiment. The La_2O_3 tungsten-alloy ITER prototype alloy was used. The exposures were primarily near the outer strike point and private flux region. Visual inspection of the exposed tungsten sample showed a uniform redeposited layer, presumably of carbon, in agreement with the previous DiMES measurements which showed partially detached divertor plasmas to have net redeposition at the outer divertor leg. Auger spectroscopy confirmed a uniform deposition of mostly carbon at a rate of about 2 nm/s. No arc traces were found on the sample, in agreement with previous DiMES experiments exposed to detached plasmas.

CONFERENCES/MEETINGS

1. A DiMES program status was presented at the High Heat Flux Components and Plasma Surface Interactions workshop in Japan in October, 1998.

PUBLICATIONS/REPORTS

1. J.N. Brooks, D.G. Whyte, “Modeling and analysis of DIII–D/DiMES sputtered impurity transport experiments,” Nucl. Fusion Vol. **39** (1999) 525.
2. D.G. Whyte, J.P. Coad, P. Franzen, H. Maier “Similarities in Divertor Erosion/Redeposition and Deuterium Retention Patterns Between the Tokamaks ASDEX Upgrade, DIII–D and JET,” Nucl Fusion Vol. **39** (1999) 1025.

Section 7

RADIATION TESTING OF MAGNETIC COIL

7. RADIATION TESTING OF MAGNETIC COIL

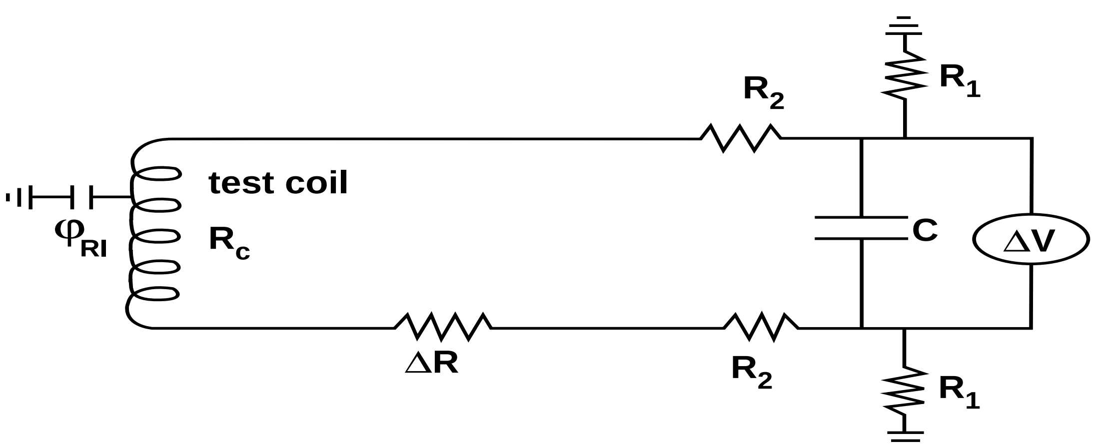
The fabrication of four prototype magnetic coils was completed and the coils were shipped to Japan Atomic Energy Research Institute (JAERI). These coils were designed, and much of the fabrication completed, in FY98. JAERI then encapsulated the coils in a Japan Materials Testing Reactor (JMTR) capsule and inserted the assembly into JMTR for radiation tests.

Upgrades to the long pulse integrator to accommodate radiation induced EMF (RIEMF) were completed. The integrator and two GA employees went to JMTR for a week of radiation tests.

The coils were irradiated in JMTR for two months. The RIEMF was measured by JAERI for all coils as a function of reactor power and time. Analysis indicates that the EMF levels do not fit any known model. In particular, the sign of the EMF changed in some cases as the power increased.

The long pulse integrator worked well. The results indicate that the drift fits the simple model (Fig. 7-1) developed from the GA FY96 High Flux Beam Reactor (HFBR) results. The integrator drift goes as

$$\begin{aligned} \Delta V_{drift} &= \varphi_{RI} R_1 \, \Delta R \big/ R_T^2 \\ &\approx I_{RIEMF} \Delta R \end{aligned} \tag{1}$$



$R_T = R_1 + R_2 + 0.5\ R_c$

Fig. 7-1. Imbalance in electrical resistance across integrator leads to drift when RIEMF present.

The drift is due to the RIEMF across the center conductor and shield, and an imbalance in the resistive loads (or perhaps the RIEMF itself) of the cable. Figure 2 shows the measured integrator drift as a function of conductor size and type. As expected, the larger the center conductor and hence the lower the resistance of the cable, the lower the drift. The integrator demonstrated acceptable levels of drift during the radiation testing, indicating that the model, and the integrator upgrade concept based on the model, are acceptable for ITER-class devices.

However, because of the unpredictability of the RIEMF, it is difficult to design coils for a fusion reactor based on the information we now have. Further understanding of the RIEMF is needed.

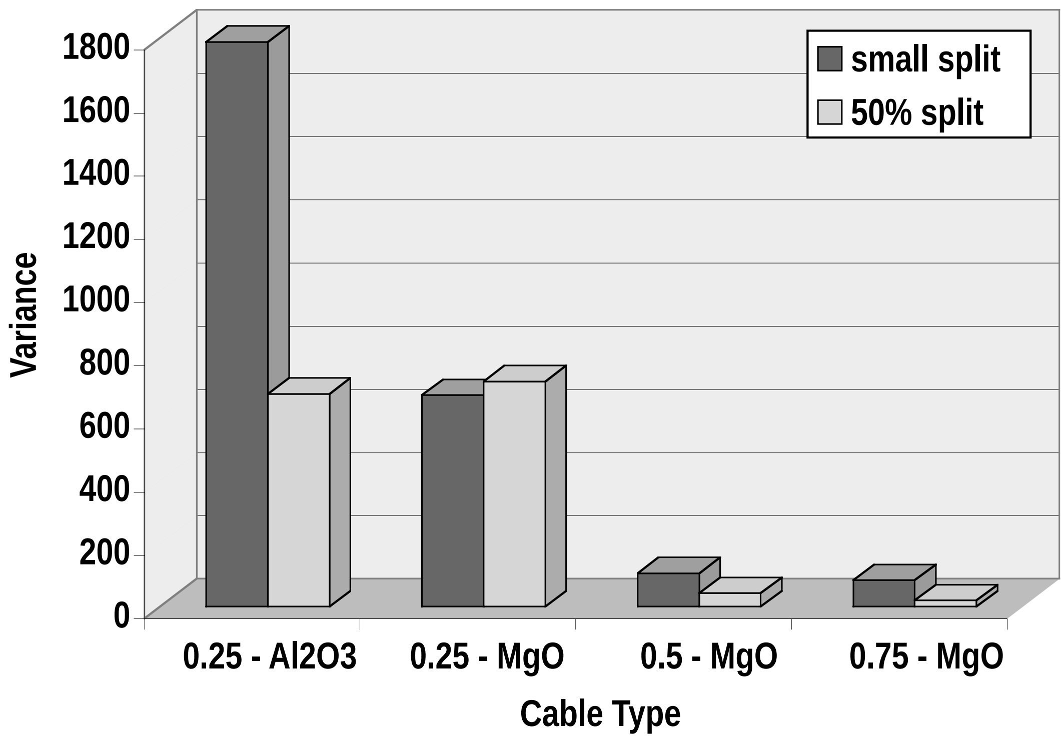


Fig. 7-2. Variance of drift by cable type and compensation method.

Section 8

VANADIUM COMPONENT DEMO

8. VANADIUM COMPONENT DEMO

A vanadium component was designed, fabricated and installed in the DIII–D vacuum vessel to determine the plasma thermal and radiation effects on vanadium alloys. The component, a small strut protection tile assembly bracket (Fig. 8-1), consists of two milled vanadium alloy plates that have been electron beam or gas tungsten arc welded. The vanadium welding techniques were developed by Oak Ridge National Laboratory (ORNL) as part of a collaboration between ORNL, GA and Argonne National Laboratory. ORNL welded the eight brackets, and inspected and radiographed each component to verify the integrity of each weld joint. A total of six projection tile bracket assemblies were installed in DIII–D. The strut protection tile assemblies were designed to protect a radiative divertor support strut located on the vessel center post. The assemblies, located in the aperture of the upper inner radiative divertor (Fig. 8-2), were mounted off the vessel center post and are indirectly exposed to the plasmas. There are two sets of three brackets located 180° apart on the vessel center post.

Tensile and impact specimens were installed adjacent to the bracket assemblies to allow for the monitoring of the parent and welded material properties without having to remove the brackets for analysis/destructive testing, . The specimens were manufactured from the same vanadium alloy parent material from which the brackets were fabricated. In addition, tensile and impact specimens were fabricated from welded vanadium material supplied from ORNL. The welded specimens include both types of weld processes, gas tungsten arc welds and electron beam welds. Over the next three years, during each of the DIII–D vents, one-third of the specimens will be removed and analyzed for the effects of radiation and thermal cycling on the structural integrity of the material and on the weld properties.

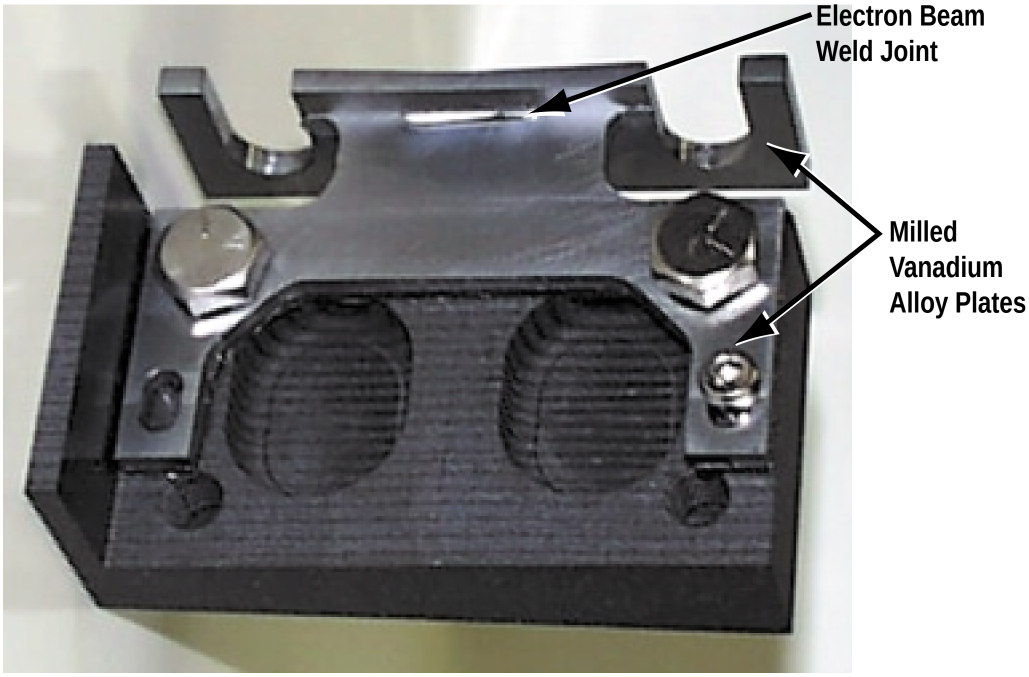


Fig. 8-1. Vanadium brackets are part of the DIII–D strut protection tile assemblies.

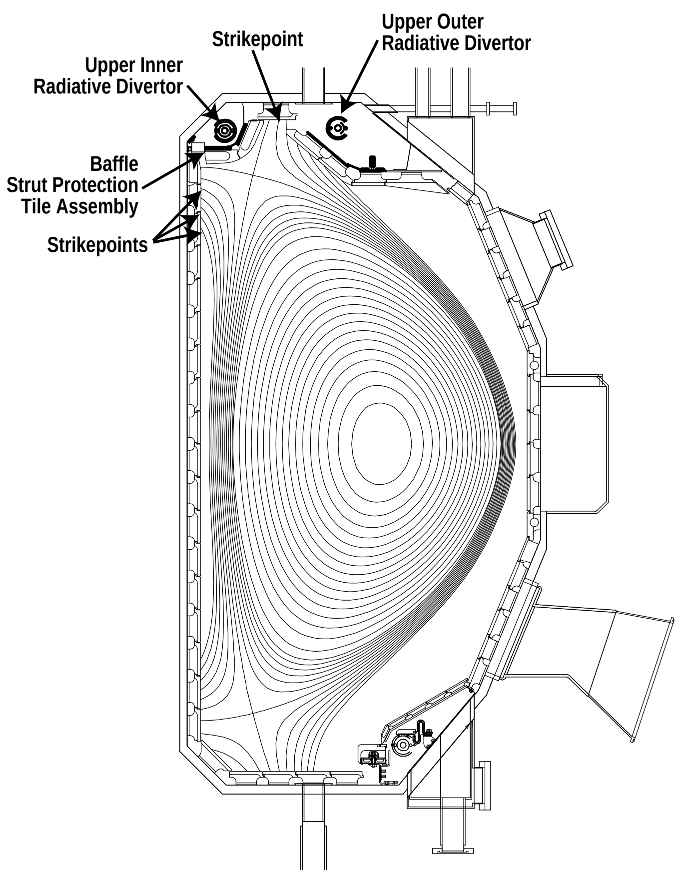


Fig. 8-2. The strut protection tile assemblies are located in the upper inner radiative divertor.

Section 9

RF TECHNOLOGY

9. RF TECHNOLOGY

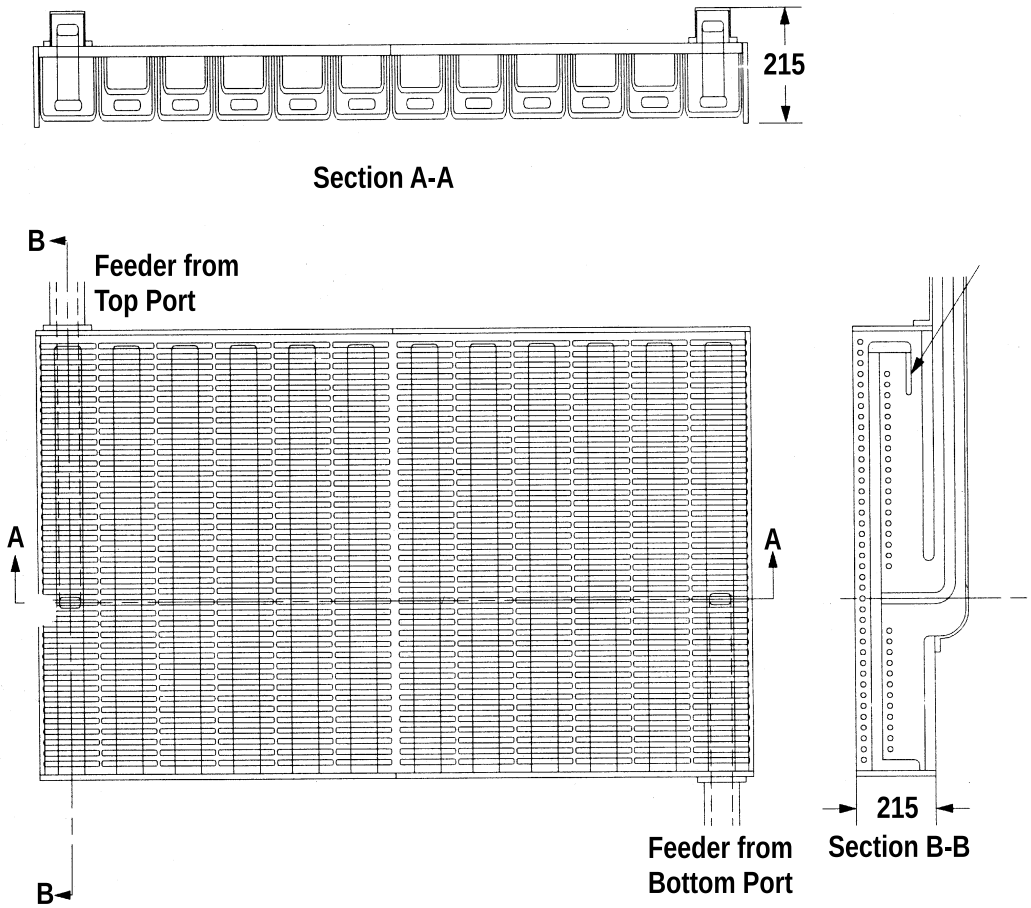
COMBLINE ANTENNA

A number of discussions were held with technical staff of Japan's National Institute of Fusion Science (NIFS) to help them in their plans for designing, fabricating and installing a combline antenna on the LHD device. In addition, NIFS has a collaboration with Dr. Takese of the University of Tokyo University to assist in the design and fabrication effort. NIFS is considering a combline antenna to be located in the high magnetic field region of the stellerator to achieve good coupling with the electrons for fast wave current drive. Initially NIFS staff considered a frequency of 35 MHz, but, based on discussions with GA staff, NIFS and Dr. Takase have decided on a higher frequency (70 to 85 MHz) to achieve better coupling. Dr. Takase has prepared a conceptual design based in large part on the GA design of the JFT-2M combline antenna. An early version of the antenna is shown in Fig. 9-1. The version shown is in a flattened configuration; later versions take advantage of the modular design of the antenna which enables the antenna to follow a helical path including a twist.

Initial discussions on the collaboration were held in Japan in October 1998. Follow-up discussions were held at the U.S./Japan RF Workshop in New Orleans in November 1998. At that time, GA obtained information on the LHD vessel design to help GA develop a conceptual design of the combline antenna. In September 1999, Dr. Charles Moeller met with Dr. Takase at PPPL while Dr. Takase was visiting there. Further discussions with NIFS staff are scheduled for early in FY00 in Japan in conjunction with the October 1999 U.S./Japan RF Workshop. GA's proposed role in FY00 and FY01 is to review the NIFS design and to participate in cold tests and possibly hot tests of the antenna.

ADVANCED ECH LAUNCHER DEVELOPMENT

An improved launcher mirror concept with enhanced heat removal capability is being developed under this task. The initial concept was based on using high thermal conductivity CVD diamond strips to remove the heat from the mirror surface to a diamond or diamond/graphite bulk material. An outgrowth of this effort was an investigation of the feasibility of using inexpensive carbon fibers rather than diamond. Carbon fibers offer the advantage of flexibility, enabling the rotating mirror to be connected by the carbon fibers, which act as a flexible heat pipe to a fixed water-cooled



Total Width 1340 mm (±9°)
Strap width 50 mm, strap sapcing 110 mm (center-center)
Faraday shield 10 mm dia., spacing 20 mm (center-center)
$k_{||} = 14.28\ m^{-1}$
$n_{||} = 19.74$ @ 70 MHz and 90° ($E_{res} = 2.7$ keV)

Fig. 9-1. LHD 12 strap combline antenna.

heat sink. The carbon fibers have lower electrical conductivity than copper, therefore, are not subject to the large disruption loads that would result if a large copper block were used as a rotatable mirror and inertial heat sink. High thermal conductivity fibers were purchased for use in testing the concept, including brazing them to graphite, various metals, and CVD diamond disks.

Preliminary thermal analyses show that a graphite mirror with a graphite fiber cooling braid connected to a water-cooled heat sink can greatly reduce the permissible time between shots compared to that expected for a radiation-cooled graphite mirror. For 5 s 900 kW pulses, a radiation-cooled graphite mirror requires approximately 15 min. cooling between shots. The addition of a graphite fiber cooling braid connected to a water-cooled heat sink reduces this time to approximately 2 min. More detailed thermal calculations need to be performed, and a prototype mirror needs to be made to confirm the predicted performance.

In a related development, GA conducted initial low power tests on its company-funded proof-of-concept remotely-steerable 110 GHz ECH launcher. The tests successfully demonstrated that steering of ± 15° can be achieved with this concept. Low power tests were also successfully carried out on this launcher with miter bends included between sections of launcher waveguide. These tests showed that the predicted steering can be achieved with or without miter bends. The “folded” waveguide has the bends out of the plane of steering. JAERI has expressed considerable interest in the concept and is interested in purchasing a prototype launcher suitable for high power tests at 170 GHz using evacuated launcher waveguide.

INTERNATIONAL COLLABORATION

A U.S./Japan RF Technology Exchange was held in New Orleans on November 12–13, 1998 in conjunction with the American Physical Society Division of Plasma Physics annual meeting. There were 10 representatives from the U.S. and 10 representatives from Japan, both from the Ministry of Education Science and Technology Administration and STA. The two-day meeting was considered beneficial by both sides, with 22 presentations and active discussions taking place. There were several mutually beneficial collaborations proposed that were subsequently included in the list of exchanges between the two countries.

The ECH International Transmission Line Workshop was held September 1–3, 1999 at Carmel, California just before the IR and Millimeter Wave Meeting held in Monterey, California the following week. Presentations were made by ECH experts from the U.S., Japan, Germany, Netherlands, Spain and Russia. Topics addressed included ECH windows and loads, corrugated waveguide and other waveguide transmission lines, quasi-optical converters and transmission lines, and matching optical units.

Plans were also made for the next U.S./Japan RF Technology Exchange. This workshop will be held at Oharai, Japan, immediately after the EC11 Conference being held in Japan October 4–8, 1999.

CONFERENCES/MEETINGS

1. 1998 U.S./Japan RF Technology Exchange, November 12–13, 1998, New Orleans, Louisiana.
2. 1999 ECH International Transmission Line Workshop, September 1–3, 1999, Carmel, California.

Section 10

IFE TARGET SUPPLY SYSTEM

10. IFE TARGET SUPPLY SYSTEM

BACKGROUND

A commercial Inertial Fusion Energy (IFE) power plant must place about 500,000 cryogenic targets each day (at a rate of 5–7 Hz) into a target chamber operating at 500–1500°C. The targets will be injected into the reaction chamber at high speed, tracked and hit, on the fly, with the driver beams. This must be done with high precision, high reliability of delivery, and without damaging the mechanically and thermally fragile targets. Key components of demonstrating a successful IFE target injection methodology are:

- Ability of targets to survive the chamber environment (target heating due to radiation and chamber gases)
- Accuracy and repeatability of target injection and tracking (ability to provide suitable beam steering and/or target steering, and shot-timing signals)
- Ability of targets to withstand acceleration into the chamber (strength of target components, including the DT itself)

The ultimate goal of this development program is to provide a successful demonstration of injecting prototypical IFE cryogenic targets into a surrogate chamber that is representative of an operating reaction chamber.

FY99 SCOPE AND OBJECTIVES

The scope of work and primary objectives of the IFE Target Injection and Tracking tasks in FY99 were as follows:

> *Review previously proposed injection and tracking methodologies and evaluate additional technologies that have become available in the 5–10 years since the major IFE design studies were conducted. Coordinate with the IFE target and plant designers as necessary. Develop an experimental plan specifying material property measurements, measurements of the thermal environment effects on filled targets, and accuracy of injection demonstrations in the laboratory.*

Brief highlights of the FY99 progress and accomplishments are covered in the following sections. Reference 1 provides a much more in-depth description.

INJECTOR DESIGN CRITERIA AND SELECTION

The technical requirements for an experimental IFE target injection system were defined, including: velocity [400 m/s direct drive (DD), 180 m/s indirect drive (ID)], acceleration (50,000 m/s^2), target spin rate (300 rev/s), target mass (5 mg DD, 1 g ID), target diameter (5 mm DD, 12 mm ID), accuracy (±0.3 mrad), tracking (target position prediction) accuracy (±0.02 mm DD, ±0.2 mm ID), injection rate (six targets in a one-second burst), and cryogenic fuel temperature limits (Phase I: room temperature injections. Phase II: 18–19 K with 0.5 K maximum temperature rise). The requirements are based on power plant injector system requirements.

A gas gun was chosen as the experimental injector over other candidates due to its expected ability to meet all requirements for target mass, velocity, acceleration, and heating – both for this experimental program and for an IFE power plant. The primary objective of the experimental program is to provide scientific data demonstrating target survivability under representative operating conditions and to provide moving projectiles to develop and test the tracking system. This objective leads to selection of a mature and simple injector technology like a light gas gun. A magnetic accelerator backup option is available. Optical tracking of the targets is planned.

TARGET HEATING CALCULATIONS

A parametric evaluation of the effects of varying the chamber gas pressure, and calculations of the asymmetric nature of target heating upon injection into a gas-filled chamber, were conducted.

The heat flux from gas-heating is related to the velocity and the pressure of the gas in the chamber. While reliable, verified models for the specific conditions of the cryogenic target in a high-temperature, reduced-pressure, gas-filled chamber are not available, the estimated total heat load (radiation plus gas-heating) is shown in Fig. 10-1. It appears that the needed heat fluxes of 1 to 2 W/cm^2 can only be achieved at very low xenon gas pressures. Injection into a chamber with the reference gas pressure of 0.5 Torr, as used in the SOMBRER0 power plant study [2], results in heat loads that are an order of magnitude higher than allowed for acceptable target heating. The gas heat load is also not uniform. Depending on the velocity of injection, the heat load at the leading edge of the target can be several times higher than the heat load near the trailing edge.

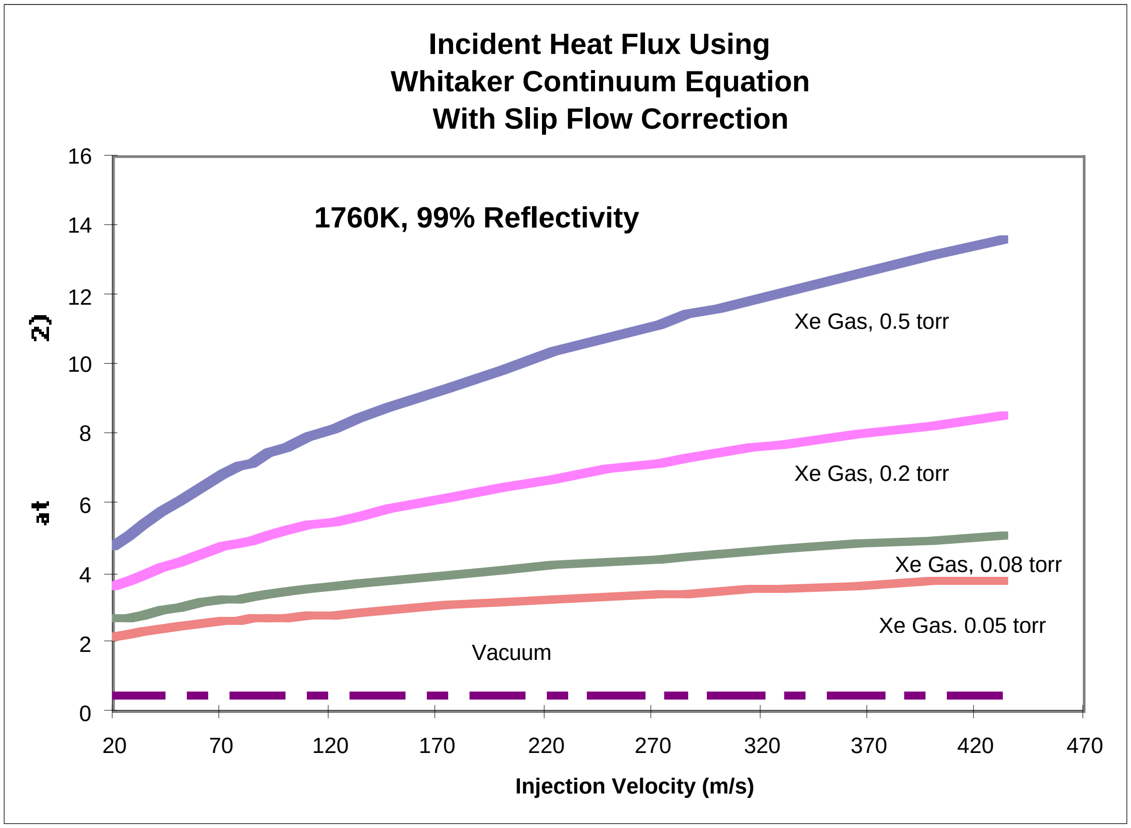


Fig. 10-1. Calculated heat flux on target under varying chamber conditions and injection velocities.

EXPERIMENTAL PLAN

An experimental plan was developed to provide data to address the key issues [1]. A series of experiments is recommended including material property measurements, measurements of the thermal environment effects on filled targets, and demonstration of the accuracy of target injection and tracking.

GA is the lead organization for target injection and tracking, and is supported by LANL. The proposed program is broken down into subtasks that follow the key feasibility issues.

The proposed sequence and logic of the experimental program is summarized in Fig. 10-2. Measurements of DT properties and initial determination of the effects of thermal radiation on filled and layered (stationary) targets takes place at LANL, concurrent with the design and initial operation of the new injection system at GA. Upgrade of the injector system to incorporate injection of cryogenically layered targets takes place in Phase II, followed by a full demonstration of injection of cryogenic targets into a hot chamber.

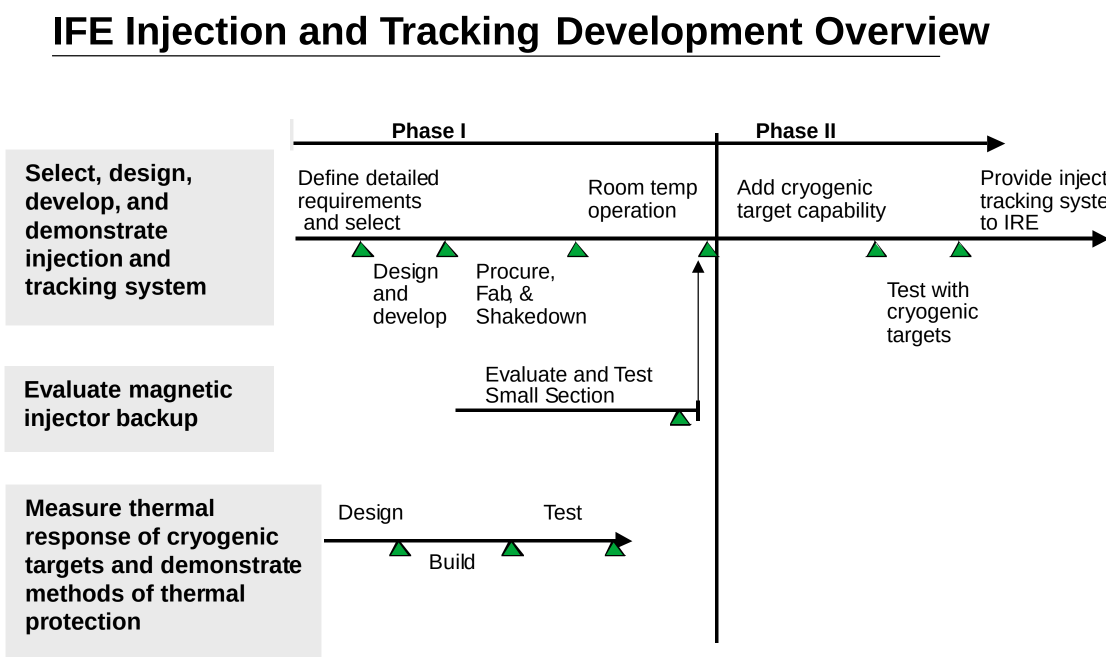


Fig. 10-2. Sequence and logic of the experimental program.

SUMMARY OF ACCOMPLISHMENTS

The FY99 accomplishments focused on several major areas:

1. Target and equipment evaluations necessary to define injection and tracking system requirements;
2. Evaluations of potential injection and tracking system technologies that could meet these requirements;
3. Selection of injection and tracking equipment for the experimental program;
4. Definition of Phase I data needs and experiments to be conducted to address the key feasibility issues of injection and tracking;
5. Meeting with target and plant designers to integrate the efforts towards practical, optimized IFE targets.

Specific FY99 accomplishments were:

1. Estimates of the effects of target chamber gas on target trajectories and target heating.
2. Calculations of the surface heat flux to the target from thermal radiation and exposure to chamber gases during injection.

3. Using a gas gun for injection, demonstration of sabot removal from a plastic capsule at room temperature with a pulsed magnet and without significant accuracy degradation (with equipment at LBNL).
4. Thermal radiation reflectivity measurements over wavelengths applicable to direct drive target chambers of gold/Kapton foils similar to those proposed for the surface of IFE direct drive targets.
5. Finite element code calculations of DT temperature distributions as a function of surface heat flux for currently proposed target designs, both direct and indirect drive.
6. Evaluations of the capabilities of a wide variety of injector and accelerator technologies, ranging from simple gas guns to advanced high-temperature superconductor systems that should become available in the next few years.
7. Specific calculations to evaluate the use of light gas guns for target injection at the low accelerations needed for IFE targets for both direct and indirect drive targets.
8. A preliminary design and thermal analysis of a gas gun sabot to protect a direct drive target from the warm propellant gases during injection.
9. Evaluations of target detection and position prediction equipment for use during injection.
10. Evaluation of the effects on indirect drive target trajectory from Flibe droplets in the injection path.
11. Preliminary feasibility calculations for using shutters and a gas counter flow to keep Flibe vapor out of the target injection area in an IFE power plant.
12. Organized the “Heavy Ion Fusion Target Workshop” and the “Direct Drive Target Workshop” to assist in coordinating efforts between groups, and to move towards practical IFE targets.
13. Participated in and made presentations at the IFE Chamber and Target Technology Meeting (March 1999) and the Snowmass Fusion Summer Study (July 1999).
14. Further development of preliminary injection and tracking system requirements for a power plant injection system, and development of requirements for the experimental injection and tracking system.
15. Development of an injection and tracking system testing and demonstration program strategy and approach.
16. Selection of recommended injection and tracking system technologies to be employed for the Phase I testing and demonstration program.
17. Recommendation of injection and tracking supporting tasks, materials data needs, and specific tests to provide that data in Phase I.

18. Preliminary recommendations of follow-on tasks to be accomplished in Phase II.

All of the above accomplishments are documented in Ref. 1.

REFERENCES

[1] Experimental Plan for IFE Target Injection and Tracking Demonstration, GA-C23241, October, 1999.

[2] Wayne Meier, et. al., OSIRIS and SOMBRERO Inertial Confinement Power Plant Designs, DOE/ER/54100-1, 1992.

CONFERENCES/MEETINGS

1. IFE Chamber and Target Technology Meeting, Pleasanton, California, March, 1999.
2. Heavy Ion Fusion Target Workshop, General Atomics, April, 1999.
3. Fusion Summer Study, Snowmass, Colorado, July 1999.
4. Inertial Fusion Energy Direct Drive Target Workshop, General Atomics, September, 1999.
5. Symposium on Fusion Engineering, Albuquerque, New Mexico, October, 1999.
6. Thirteenth Target Fabrication Meeting, Catalina, California, November, 1999.
7. Second Japan-U.S. Workshop on Inertial Fusion Energy, Osaka, Japan, November 1999.

PUBLICATIONS/REPORTS

1. “Experimental Plan for IFE Target Injection and Tracking Demonstration,” GA-C23241, October, 1999.
2. “Target Injection and Tracking for IFE,” Ronald Petzoldt, Oral presentation at the Symposium on Fusion Engineering, Albuquerque, New Mexico, October 27, 1999.
3. “Target Injection and Tracking for Inertial Fusion Energy,” Ronald Petzoldt and Dan Goodin, Oral presentation at the IFE Chamber and Target Technology Meeting, Pleasanton, California, March 18–19, 1999.
4. “Target Fabrication, Injection and Tracking,” Ken Schultz, Gottfried Besenbruch, and Ronald Petzoldt, Oral presentations at the Fusion Summer Study, Snowmass, Colorado, July 1999. Also contributed to the Fusion Summer Study final report.
5. “Status of Target Injection and Tracking Studies for Inertial Fusion Energy,” Ronald Petzoldt, Dan Goodin, and Nathan Siegel, Poster presentation at the 13th

Target Fabrication Meeting, Catalina, California, November 8–11, 1999. Paper submitted to *Fusion Technology*, November 1999.

6. Minutes of the “Heavy Ion Fusion Target Workshop”, General Atomics, ICFT99/120, April 22, 1999.
7. Minutes of the “Inertial Fusion Energy Direct Drive Target Workshop,” General Atomics, ICFT99/227, September 15, 1999.
8. “IFE Target Injection and Tracking,” Presented by Dan Goodin at the Second Japan-U.S. Workshop on Inertial Fusion Energy, Osaka, Japan.
9. “Mass Production of Inertial Fusion Energy Targets,” Presented by Dan Goodin at the Second Japan-U.S. Workshop on Inertial Fusion Energy, Osaka, Japan.
10. “IFE Target Fabrication and Injection,” Ken Schultz, General Atomics, Presentation to the Secretary of Energy Advisory Board (SEAB), 26 May 1999.
11. “The Inertial Fusion Energy Program,” Ken Schultz and Wayne Meier, Oral presentation at the 13th Target Fabrication Meeting, Catalina, California, November 8–11, 1999.