MEETING REPORT

SUMMARY OF THE 14TH IEEE/npss SYMPOSIUM ON FUSION ENGINEERING, SAN DIEGO, CALIFORNIA, OCTOBER 1-3, 1991

INTRODUCTION

Over the past 27 yr, the Institute of Electrical and Electronics Engineers (IEEE)/nuclear plasma systems society (npss) Symposium on Fusion Engineering has become a leading technical forum for exchange of information on the engineering and technology of fusion energy. The 14th symposium was held October 1-3, 1991 in San Diego, California. This report includes a brief summary of each technical session written by the session chairpersons. The symposium proceedings, including a full paper on each technical talk, was published in January 1992 by the IEEE.

OPENING SESSION

The National Energy Strategy and Fusion
J. F. Decker (DOE)

The development of fusion energy is an important element of the U.S. national energy strategy, described by J. F. Decker [U.S. Department of Energy (DOE)]. Research and development programs (R&D) for both magnetic and inertial fusion energy are included. Key elements of the magnetic fusion energy program are as follows:

1. maintenance of a core physics and technology program
2. performing deuterium-tritium (D-T) experiments in the Tokamak Fusion Test Reactor (TFTR)
3. achievement of a burning plasma experiment
4. participation in the International Thermonuclear Experimental Reactor (ITER) activity
5. development of a materials testing facility
6. operation of a steady-state tokamak experiment.

Key elements of the inertial fusion energy (IFE) program build on the inertial confinement fusion (ICF) portion of the DOE defense program:

1. achieving high gain in the Nova Upgrade
2. operation of the Laboratory Microfusion Facility (LMF).

The IFE portion of the DOE energy research program would then add the following:

1. development of a heavy-ion beam driver
2. operation of the Engineering Test Facility.

These elements would lead to a demonstration fusion power reactor by the year 2025. Fusion energy is an important element of the national energy strategy, despite the long-range payoff for fusion research, because of the continuing belief that fusion will become a safe, clean, economic, and practical source of energy for humankind in the future.

Status of the U.S. Magnetic Fusion Program
N. A. Davies (DOE)

N. A. Davies, director of the DOE Office of Fusion Energy (OFE), informed the symposium attendees of the recent changes in the fusion program in response to directions from the Secretary of Energy. Secretary Watkins appointed a Secretary of Energy Advisory Board (SEAB) to give him advice regarding the priorities of projects in energy research in light of current plans to keep the total funding for Office of Energy Research fixed at the FY1990 level for ~5 yr. The SEAB recommended that fusion must maintain scientific momentum and that the fusion budget must rise at ~5% real growth per year. While this implies a high priority for fusion within energy research, this budget was deemed inadequate to continue a vigorous effort toward a burning plasma experiment, and it was recommended that primary emphasis should shift to a strong U.S. participation in the ITER program. The SEAB recommended that the fusion program define a new experiment in the $500 million class to study tokamak improvements.

The Director of the Office of Energy Research, W. Happer, appointed a Fusion Energy Advisory Committee (FEAC) to advise him of the impact of these recommendations on the fusion program. The FEAC recommended that the Burning Plasma Experiment (BPX) be deferred in favor of full and vigorous U.S. participation in ITER. Thus, ITER will become the main thrust of the U.S. magnetic fusion program. Funds currently earmarked for BPX should be used to strengthen the base program and to plan initiatives to fill the gap between the end of TFTR and the start of operation of ITER.

Davies emphasized that although the decision not to proceed with BPX was very painful, the fact that fusion could hope for modest funding growth in the current U.S. budget situation was very positive. The fusion program is now faced with the challenge of how to best use the resources planned for BPX in FY1992 in order to move vigorously forward in FY1993 with new initiatives such as upgrades of existing experiments or possibly new experiments that could be done within the budget constraints. All participants in the fusion program were urged to give this challenge their best efforts.

PLENARY SESSIONS

The International Thermonuclear Experimental Reactor
P. A. Politzer (GA)

A. Glass [Lawrence Livermore National Laboratory (LLNL)] reviewed the status of the ITER project in his paper, "The ITER Activity." Final approval of the international agreement for the engineering design activity (EDA) is expected early in 1992. The agreement covers the EDA sites (San Diego, Garching, and Naka), the management organization, and the framework for relations among the parties. The interaction between the Joint Central Team and the four parties will be on the basis of formal task assignments in three areas: technology R&D, physics R&D, and design. The parties will bid for specific assignments, and on the basis of these bids, the director will prepare a work plan to be submitted to the ITER Council.

The U.S. ITER home team organization is in place, with managers selected in the areas of engineering, technology, and physics. The DOE has established an interim ITER organization in the OFE. Although the basic agreement has been worked out among the four parties, discussions are continuing on many items, such as the responsibilities of the three EDA sites, division of responsibility between the Joint Central Team and the home teams, and budget issues.

As soon as the EDA agreements are approved, two special working groups will develop the technical objectives of the EDA in support of the ITER mission, and they will establish guidelines for the implementation of task assignments. The ITER work plan will be completed in this period.

Finally, Glass noted that the world view of ITER has changed significantly. The project, and fusion as a whole, has much more visibility and has received a more positive response. There is more acceptance that ITER is a machine that will be built. It is being viewed as a prototype for international projects for the 21st century.

TFTR and JET Status
J. L. Luxon (GA)

The two invited papers emphasized the plans of two major tokamaks to operate with a D-T mixture. The paper "JET Status and Prospects," presented by M. Huguet [Joint European Torus (JET)], addressed plans to go forward with D-T operation in 1995, along with plans for a proof-of-principle experiment in late 1991 using a small amount of tritium beam injection into deuterium plasmas. The paper "TFTR Recent Results and Preparation for D-T Experiments," by R. Hawryluk [Princeton Plasma Physics Laboratory (PPPL)], addressed plans of the TFTR group to begin D-T operation in 1993. The planned experiments on these two devices will provide the first opportunity to assess confinement in tritium, evaluate heating by alpha particles, evaluate alpha-particle collection effects, and gain operating experience with tritium systems.

The JET team achieved their highest value of $n_e(0)t\ T_i(0) = 9 \times 10^{20}$ keV/s$\cdot$m^{-3} in 1990 using a divertor plasma configuration. In the past year, they demonstrated 60-s-long, 2-MA discharges using 4 MW of ion cyclotron resonance heating (ICRH) power for heating and lower hybrid power to provide partial current drive. In addition, one cycle of alternating current (ac) tokamak operation was demonstrated to offer a potential alternative for a fusion reactor. Successive 10-s-long pulses were demonstrated at 2 MA with dwell time as little as 50 ms.

JET operation with beryllium walls has improved plasma performance and reduced impurity influx as compared with earlier carbon configurations. However, recognizing that their best performance is still transient, with the high performance maintained for <1 s, the JET team is now planning a quasi-steady-state pumped divertor configuration with four new coils inside the vessel. It is anticipated that this phase will begin in 1993, and the JET program will then be extended through 1996.

The TFTR device is returning to operation after being shut down for ~1 yr. Earlier results were summarized in which values of $n_e(0)t\ T_i(0) = 4.3 \times 10^{20}$ keV/s$\cdot$m^3 and $Q_{DD} = 1.9 \times 10^{-3}$ were achieved. This leads to a calculated equivalent value of $Q_{DT} \cong 0.38$.

Extensive preparations are under way for D-T operation in TFTR. A number of new tools are being developed, including a tritium pellet injector, specialized wall conditioning techniques, and additional ion cyclotron resonance frequency (ICRF) power. Shielding of diagnostics is being carried out, and specialized alpha-particle diagnostics are being developed. Detailed analysis of the site radiation has been carried out. Tritium retention in carbon wall tiles is a major concern and could limit D-T operation because of site inventory limitations. Careful assessment of this issue is ongoing, and laboratory measurements of retention in representative tiles are being performed.

The Burning Plasma Experiment
R. Stambaugh (GA)

J. Sinnis (PPPL) presented "The BPX Project," a review of the BPX project and device. He announced that about a week prior to the symposium, the DOE had determined that funding constraints projected through 1996 would make it impossible to build BPX. He pointed out that the current BPX design supported its mission and that its careful cost analysis would be a benchmark for future initiatives to build major fusion devices.

The mission of BPX included determining the behavior of self-heated plasmas, demonstration of production of substantial amounts of fusion power (in excess of 100 MW), investigation of key features of alpha-particle physics, and gaining experience in power production and handling D-T fuel. The BPX program would allow ITER to focus on long-pulse, high-fluence plasmas.

The BPX device is a step of a factor of ~10 over current devices, an appropriate intermediate step to ITER. The fusion power output would be in the 100- to 500-MW range. The expected energy gain Q is 9 with a one standard deviation range of $3 < Q <$ ignition. The machine is liquid-nitrogen-cooled copper. The divertor is pyrolytic graphite with separatrix sweeping to hold the surface temperature of the graphite under 1700°C. The basic machine is built in six sectors for remote maintenance. The toroidal field (TF) coil is a beryllium-copper alloy with a peak stress of 700 MPa for a 9-T field in the plasma. The poloidal field (PF) coil system supports 11.8-MA plasma operation in both single- and double-null divertor configurations. The heating system is ICRF in the 60- to 90-MHz range. Plasma diagnostics will have the added challenge of high neutron fluxes and remote maintainability. Extensive alpha-particle diagnostics are planned.

The BPX project had an international conceptual design review in the spring of 1991. The panel concluded that the achievement of $Q > 5$ was a conservative expectation. They found the engineering to be well conceived and a sound basis for detailed design. The cost estimate was judged to be accurate overall. The environmental assessment was expected to reach a finding of no significant impact. First plasma was scheduled for the year 2000 with the D-T phase beginning 2 yr later.

Inertial Confinement Fusion
M. Monsler [W. J. Schafer Associates (WJSA)]

In his plenary talk "ICF Results and Plans," M. Sluyter (DOE) discussed the Office of Inertial Fusion program. The ICF program is funded at $171 million in FY1991 and $190 million in FY1992, including funds for upgrading the OMEGA laser at the University of Rochester. The four important milestones of the ICF program are as follows:

1. *scientific feasibility*: already demonstrated in underground experiments in the Halite/Centurion program
2. *ignition and gain*: currently the goal of the Nova and OMEGA Upgrades to come
3. *defense applications*: using the LMF to obtain high yields for weapons physics experiments and weapons effects testing
4. *electrical power*: using rep-rated drivers and reactors in an electricity-producing demonstration power plant.

Sluyter said that the decision to build the Nova Upgrade had been deferred a year or two because of the budget implications of the estimated construction cost of $350 to $400 million. The current ICF program emphasizes three major points:

1. precision Nova (1992)—implosions and hohlraum experiments with pulse shaping and beam balancing
2. hohlraum laser physics campaign on Nova (1994)—to improve the radiation environment necessary to drive a fuel pellet symmetrically and efficiently
3. hydrodynamically equivalent physics campaign on Nova (1994)—to address capsule implosion physics issues of stability, mix, and hot-spot ignition.

Finally, Sluyter discussed the crucial nature of the current light-ion beam-focusing experiments on PBFA-II at Sandia National Laboratories (SNL). Their goal is to reach ≥ 10 TW/cm^2 on target to show adequate drive power density on hohlraum targets.

JT-60 Upgrade Status
R. L. Freeman (GA)

T. Kimura [Japan Atomic Energy Research Institute (JAERI)] presented a paper titled "Initial Results from the JT-60 Upgrade." This paper reviewed the objectives of the JT-60U device and described the experimental results to date. Initial operation with JT-60U achieved a 4-MA divertor plasma with ohmic operation in April 1991. The best results to date include stable 4-MA neutral-beam-heated discharges with stored energies up to 5.1 MJ. In addition, deuterium-deuterium (D-D) neutron emission rates of 1.3×10^6/s were achieved in a "hot-ion" mode. Studies of the magnitude and scaling of heat flux on the first wall were completed, and ripple losses were observed. Recently, emphasis has been placed on the wall conditioning that has led to Z_{eff} values of ~3. Kimura also described the JT-60 control systems in some detail.

DIII-D Results
R. J. Hawryluk (PPPL)

T. Simonen [General Atomics (GA)] gave a plenary talk on "Recent Results from DIII-D and 1990's Research Plans." The long-range goal of the DIII-D program is the attainment of 2 MA of noninductive current at $\beta = 5\%$ for 10 s. The successful achievement of high-beta discharges is being followed up with comprehensive current drive and divertor programs. The highest value of average beta achieved to date is 11%, with central values of $\beta(0) = 44\%$. Analysis indicates that the q profile is hollow (i.e., negative shear in the core), and the core is in the second stability region while the edge is in the first stablility region. In addition to neutral beam and bootstrap current drive, the DIII-D group is investigating electron cyclotron heating (ECH) and fast wave. The advanced divertor program is studying the effects of divertor bias that indicate it is possible to control the density and build up a substantial neutral pressure in the baffle region. Of considerable interest was the report of the achievement of enhanced H-mode confinement (VH mode) following boronization, with energy confinement almost four times that of L mode.

ORAL SESSIONS

There were three parallel oral sessions each morning and afternoon. The 18 sessions are summarized here.

The ITER Project
P. A. Politzer and F. Puhn (GA)

An overview of the ITER project technical progress through the conceptual design activity (CDA) was presented by three invited speakers. The ITER project has completed a critical phase and is currently being organized to enter the EDA phase.

In "Physics Aspects of the ITER Design," D. Post (PPPL), the U.S. ITER team physics leader, reviewed the physics issues that led to the CDA design and discussed the process of selection of the ITER parameters. The physics mission of ITER is the achievement of long-pulse ignited operation with the ultimate goal of steady-state operation. The guidelines for the physics design were based on the experience base of tokamak research, including energy confinement data, from which the ITER team developed scaling relations, and rules for operational limits on plasma beta, safety factor, and density. In examining the CDA design, the primary areas of concern are the high heat fluxes at the divertor and the very high peak heat loads during disruptions. There are problems with heat loads and erosion.

The engineering design concept for ITER was presented by J. Doggett, the U.S. ITER team engineering manager. The conceptual design of ITER is impressive in its level of detail. Many challenging engineering problems remain, such as heat removal and reacting disruption loads.

C. Baker, the U.S. ITER team technology manager, presented the status of the ITER technology R&D program in support of the EDA. Current planning is based on the R&D plan as developed by the CDA, which calls for approximately $800 million worth of R&D by the four parties during the EDA. One result of the U.S. assessment of the CDA was that this R&D plan has some inadequacies. It does not address reliability; it needs more emphasis on innovative solutions to problems in areas such as divertors; and insufficient attention is given to safety and environmental issues.

Power Systems
A. Nerem (GA) and D. J. O'Neill (PPPL)

Three scheduled presentations (those from Instituto Gas Ionizzati and University of Texas) were withdrawn, leaving only the invited speaker, D. Huttar (PPPL), who discussed alternatives for powering the TF, PF, and field-shaping coils of the BPX reactor. Common to all the alternatives was the use of the current TFTR power system. The alternatives included an upgraded utility grid with synchronous machine interface, a new flywheel/alternator, and batteries as energy storage for pulsed loads. The selected system approach utilized a synchronous machine interface to provide VAR compensation and accelerated energy extraction, thereby maximizing the contribution from the utility grid. The paper provided an excellent overview of the problems and alternative solutions for powering the large tokamaks of the future.

Vacuum, Divertor, and First Wall I
M. J. Schaffer (GA)

Basic vacuum technology was addressed in "Helium Self-Pumping by Tokamak Pump Limiter Materials Exposed to an Electron Cyclotron Resonance Plasma," by C. A. Outten et al. (University of Michigan). Freshly deposited single- and multiple-layer coatings were exposed to helium-containing simulated tokamak scrape-off plasmas from an electron cyclotron resonance source, and the helium retention at various sample temperatures was measured. Nickel coatings exhibited better helium retention over the range of test conditions than either vanadium or tungsten.

"Structural Design of the Tore Supra Phase III Limiter Head," by J. F. Dempsey et al. (SNL), discussed the thermomechanical design of the planned phase III pump limiter head for Tore Supra. The limiter head consists of graphite-tile-armored tubes cooled by poloidally flowing pressurized water. Each tube was initially mechanically independent from the others. However, unequal heating and the consequent unequal thermal expansion made the hotter tubes rise and intercept still more incident plasma power. Various means of linking the tube assemblies were studied, and a novel wire-weave linkage was chosen for this purpose.

Two vessel/divertor/first-wall system designs were presented. R. E. Rocco (Ebasco) presented "An Overview of the Burning Plasma Experiment (BPX) Vacuum Vessel System," a conceptual design that with the demise of the BPX project will not be developed further. B. LaBombard [Massachusetts Institute of Technology (MIT)] presented "Design of Limiter/Divertor First-Wall Components for Alcator C-Mod," which has been built but not yet operated. Both tokamaks use thick-wall Inconel plasma chambers to resist large disruption electromagnetic forces. Both apply inertially cooled tiles over the full first wall and divertor. Graphite tiles were chosen for BPX because of their proven performance at high heat flux in a tokamak environment. Molybdenum tiles are used in C-Mod because they condition much more rapidly than graphite at the maximum allowable C-Mod baking temperature of 150°C and because molybdenum has served well in this role in previous Alcator tokamaks. The C-Mod tile assemblies have been designed to withstand all disruption forces. However, the BPX tile assemblies rely on the assumption that electric currents can be limited by substrate insulation. This seems risky because current can flow between tiles through plasma along magnetic lines, permitting large-scale currents and destructive forces.

While there has been progress in the design and construction of tokamak plasma chambers, first walls, divertors, and limiters able to absorb large heat and particle fluxes, the current designs still do not extrapolate with confidence to a fusion reactor. In the designs presented in this and other sessions of the conference, the high heat flux can be handled only by mounting a large number of small pieces on flexible, relatively fragile supports cooled by intricate in-vacuum water circuits. Furthermore, all these systems must be maintained remotely. The resulting armor is extremely vulnerable to destruction by disruption electromagnetic forces. The solution to the disruption problem might have to be absolute disruption avoidance rather than ultrarugged components. Alternatively, paths to new design concepts must be opened. For example, if there were refractory, high-thermal-conductivity, gas-impermeable, plasma-compatible composites and the accompanying manufacturing technologies to permit the construction of complete, electrically nonconducting plasma chambers and/or divertors, such components would be free from disruption forces. Then, their design would only have to contend with the still difficult thermal and erosion requirements.

Recent Experimental Results
A. P. Colleraine and P. A. Henline (GA)

This session was devoted to talks on recent hardware activities of the major tokamak experiments: Tore Supra, JET, JT-60, and the Microwave Tokamak Experiment (MTX). A common theme running through the session was the increasing emphasis on disruption physics and the problems of divertors in handling the required particle fluxes and heat loads. The development of the new actively pumped limiter/divertor concept at Tore Supra shows the technology progression required in this first-of-a-kind superconducting tokamak where long-pulse operation is routine. The inner carbon wall in this machine is able to remove 12 MW of heat continuously, and the actively cooled limiter is able to handle peak heat loads of 10 MW/m^2 (or 3.6 MW/m^2 average).

The JT-60 machine is now undergoing startup following its extensive modification program to obtain higher plasma currents with single-null divertor operation. The machine is now capable of using the full 4.2-T toroidal fields, a flux swing of 61 V/s for a 6-MA plasma, and 40 MW of neutral beam heating. Plasma disruption data have been obtained, and the current change per second seems to rise linearly up to 800 MA/s at 4 MA. The most severe limitation on future operation appears to be the divertor heat handling capability. Peak values of energy dumped of 30 MJ are obtained with 80 MJ input from beams and joule heating.

Novel operation of the JET tokamak as an ac machine was described. It has been possible to sustain an initial plasma with the conventional biasing of the ohmic heating circuit, then ramp this plasma down to zero (at up to 2 MA/s), swing through zero, and reestablish an essentially identical plasma in the opposite bias direction. Some minimum dwell at zero crossing is found to be needed to obtain the correct chamber prefill pressure for good breakdown. In addition, error fields need to be well controlled, and plasma vertical stability is poor at currents below ~80 kA. The interesting subject of halo currents in JET was also addressed. The currents induced in wall structures have long been seen to cause major damage in tokamaks when they flow through elements that are not designed to take the substantial loads that arise. Examples of arcing between components and the bending and tearing of wall tile supports were shown. General prescriptions for avoiding such problems were given, and the importance of not insulating the first wall from the vacuum vessel was stressed.

The state of the art in ECH technology was addressed in the MTX experiment. The heat flux problems in current gyrotron designs becomes intolerable at frequencies of much above 140 GHz, and it is for this reason that the LLNL group started their experiments with a free electron laser (FEL) system. This, however, may only be trading one set of problems for another. The technological challenges in aligning and operating a 6-MeV FEL at a repetition rate of only 0.5 Hz with 20-ns pulses were an education to us all.

Plasma Engineering and Particle Control
J. A. Wesley (GA)

G. Neilsen [Oak Ridge National Laboratory (ORNL)] presented an invited paper on the BPX plasma engineering. The presentation described recent plasma engineering improvements implemented in the BPX design. A larger divertor area allows more room to sweep the separatrix strikepoint and extends the pulse-length capability of BPX from 3 to 7 s. The accuracy required for the PF coil systems has been addressed using a model that quantifies the effect of such errors on the magnetic island structure of the plasma. Avoiding locked modes and bifurcation of the divertor heat load sets a very stringent tolerance, on the order of a few millimetres, on the accuracy and positioning of the PF coils. This accuracy is likely beyond that achievable during fabrication and assembly; therefore, error-correcting coils to fine-tune the island structure are incorporated as part of the BPX design.

M. J. Schaffer (GA) presented an invited paper that described the results obtained in experiments with an electrically biased baffle added to the lower divertor of the DIII-D tokamak. In experiments without an applied bias, neutral gas pressures of up to 10 mTorr were obtained beneath the baffle when the separatrix was optimally positioned. With bias applied, similarly high pressures were obtained with the separatrix far from the baffle entrance aperture. This effect is explainable in terms of $E \times B$ flows driven in the plasma scrape-off layer. Advanced divertor operation also resulted in reduced plasma core density and increased plasma core temperature during H mode. Future experiments will add active pumping in the baffle plenum for plasma density control during radio-frequency (rf) current drive experiments.

A. Tanga (JET) presented a study of plasma disruptions in JET. The effect of plasma elongation and wall material on the characteristics of disruptions in JET were described. For a given wall material, the rapidity of the current quench increases with both the plasma current level and the initial elongation of the plasma. With carbon walls, current quench rates with high-current, high-elongation plasma exceeded 300 MA/s. The highest quench rates were associated with an accompanying rapid vertical instability owing to loss of vertical position control. With beryllium-coated walls and limiter, quench rates were much lower, ~5 MA/s, provided the plasma position could be stabilized. In cases where stabilization is lost, the quench rate again increases to high levels. In either case, vertical instabilities result in large forces on the vacuum vessel that pose a serious operational concern. A disruption mitigation system has been put into operation that uses sensing of the disruption precursors to trigger a rapid decrease in elongation prior to the onset of the current quench. This strategy has been successful in preventing large vessel forces in operational regimes where disruptions are difficult to avoid.

S. K. Ho [University of California-Berkeley (UCB)] described theoretical and scoping studies of the feasibility of using compact toroids (CTs) for efficient central fueling of the ARIES-III D-^{3}He power reactor. Conventional fueling pellets and gas injection are expected to be ineffective owing to the high temperature of the ARIES-III plasma. Fueling with CTs appears to offer the only possible means of fueling deep into the plasma core. Injection power requirements are high enough to have an appreciable impact on the reactor electrical recirculating power.

M. Firestone (Mission Research) described work on "Comprehensive Control of a Tokamak Reactor," using linear optimal control theory to develop a control system for future reactors. Much of the work to date has focused on methods development and on the magnetic control aspects, but the method also contains provisions to incorporate plasma parameters such as temperature, density, and resistivity into the control system.

Fusion Power Reactors I
S. Davis (PPPL) and C. Baker (ORNL)

F. Najmabadi [University of California-Los Angeles (UCLA)] opened the session with a description of the ARIES-III design, and C. G. Bathke [Los Alamos National Laboratory (LANL)] followed with a talk titled "The ARIES-III D-^{3}He Tokamak Reactor: Design-Point Determination and Parametric Studies." The ARIES-III design requires higher temperatures ($T_i \cong 50$ keV) over the conventional D-T approach, with betas in the second stability regime. The design also requires an energy confinement enhancement factor of ~7 over the ITER89-P scaling. Bathke discussed variations in the cost of electricity (COE) with design and operational parameters. The reduced neutron flux with D-^{3}He results in a more manageable level of radioactive byproducts. This is clearly the main benefit of the D-^{3}He approach.

M. S. Tillack (UCLA) presented "Design and Analysis of a Wetted Wall ICF Reactor Cavity." A thin lead film was fed to the cylindrical cavity wall from a porous silicon carbide (SiC) composite structure. The lead coolant enables removal of roughly one-quarter of the heat to the first wall. Since rapid pulsing is expected in a reactor, the cavity clearing time for the lead to condense back to the wall will be critical.

W. J. Hogan (LLNL) discussed design studies in relation to "Small Inertial Fusion Energy (IFE) Demonstration Reactors." The target ignition "cliff" can be shifted to lower energy with the penalty that the gain achieved at a given drive energy is also smaller. Using these targets, smaller, less expensive facilities could be built. These facilities could include a low-cost single-pulse experiment, advance to a burst-mode facility (2 to 100 shots at 1 to 10 Hz), and then to longer duration tests (100 to 10 000 shots).

V. D. Lee (McDonnell Douglas Missile Systems) presented "Design Integration of an Inertial Fusion Energy Reactor Power Plant," the design of an IFE reactor with a 5-MJ KrF excimer laser driver. The device is able to produce roughly 1000 MW of electric power from a gross output of 1400 MW. The design included an overview of the complete plant, including the cavity, blanket, 60 symmetrically arranged beamlines, a tritium processing facility, steam generators, and other support facilities. The design included the lead-wetted wall ideas discussed earlier by Tillack.

Next-Generation Designs
R. L. Miller (LANL) and J. A. Leuer (GA)

P. A. Politzer (GA) described the design of a steady-state tokamak experiment that might begin operation in 1999. Such a machine would emphasize physics issues related to efficient current drive and high bootstrap current, profile control, and impurity control. Also, important technology issues related to high heat flux and steady-state operation would be addressed in a superconducting magnet configuration with a nonburning (possibly hydrogen) plasma. A similarly motivated paper was presented by L. Pieroni (Frascati) describing the conceptual divertor optimization and steady current study (DIOSCUR). This machine has major plasma parameters of $R = 3.7$ m, $A = 3.5$, $I = 10$ MA, and $B = 7$. T. M. Nassi (MIT) summarized a design for the Ignitor D-T compact ignition experiment, emphasizing high field (13 T) and predominantly ohmic heating. A comparison of ignition conditions based on a zero-dimensional plasma model for tokamaks and reversed-field pinches (RFPs) was presented by C. G. Bathke (LANL) on behalf of K. A. Werley (LANL). This study indicated the RFP configuration contains a number of favorable properties over the tokamak configuration and could lead to a simpler and less expensive ignition device.

Papers describing a variant of the ITER design at a higher plasma aspect ratio of 4 and corresponding physics design guidelines were presented by L. J. Perkins (LLNL) and N. A. Uckan (ORNL), respectively. Operation at higher aspect ratio is seen to result in reduced plasma current requirements, longer inductive burn capabilities, higher bootstrap current contributions, eased divertor performance, and higher neutron fluences in the ITER technology phase. Typical ignition plasma parameters for the high-aspect-ratio design (HARD) are $R = 6.3$ m, $A = 4.0$, $I = 15$ MA, and $B = 7.1$ T, in contrast to the CDA parameters of $R = 6.0$ m, $A = 2.8$, $I = 22$ MA, and $B = 4.9$ T. The device is more reactor relevant and will provide the high-fluence technology testing needed for a fusion power demonstration (DEMO) facility.

J. C. Commander [Idaho National Engineering Laboratory (INEL)] described the layout of the conventional facilities for the proposed BPX. An important theme of the proposals for experiments described in this session is the potential attractiveness of tokamak operation at higher plasma aspect ratio, as suggested by the ITER HARD study and recent Japanese Steady-State Tokamak Reactor (SSTR) and U.S. ARIES-I commercial reactor designs, described in other sessions at this symposium.

Vacuum, Divertor, and First Wall II
L. Sevier (GA)

This session described divertor and first-wall requirements for current devices and those beyond ITER. The BPX uses a carbon-based divertor with inertial cooling. The divertor, described in papers by J. Haines [McDonnell Douglas Aircraft Corporation (MDAC)], M. McSmith (PPPL), and R. Langley (PPPL), is designed for a 5-s operation, 1-h cooldown, and a maximum graphite temperature of 1700°C. Various active cooling methods were investigated but were not found significantly more effective than inertial cooling for the relatively short pulse. Annealed pyrolytic graphite was selected as the divertor plasma facing material. A single-pass separatrix sweep was adopted as the baseline method of limiting the divertor surface temperature. Key issues remaining for the design are disruption resistance, pyrolytic graphite delamination, and remote alignment methods.

The JET advanced divertor will be a cryopumped design, described by M. Huguet (JET). The magnetic configuration is achieved by manufacturing four water-cooled coils internal to the vessel. This fabrication process is expected to take 4 to 5 months. A two-stage upgrade process will be used for the divertor. The first stage uses radiation-cooled beryllium tiles, with a limit on deposited power. The second stage will use beryllium tiles with hypervapotron cooling to provide steady-state operation at 40 MW. The separatrix will be swept at 4 Hz over a 20-cm distance. The tiles remain below 700°C for up to a 12 MW/m^2 time-averaged load. The primary design issue is development of the technique to bond beryllium to the copper cooling channels so that delamination will not occur.

Papers by C. P. C. Wong (GA) and I. Sviatoslavsky (University of Wisconsin), described the conceptual design of the divertor and first wall for the ARIES-III 1000-MW(electric) tokamak reactor using the D-^{3}He fuel cycle and operating in the second stability regime. A conventional tungsten divertor operating with a 620°C surface temperature is used with resulting negligible net erosion. The first wall is designed with a low-activation ferritic steel structure with a beryllium first surface coated on a thin tungsten diffusion barrier. Beryllium provides the high electrical conductivity needed for reflection of synchrotron radiation and acts as a sacrificial melt layer.

Safety considerations are of growing importance in both current and future fusion machine design. Plasma-facing components may dominate the safety performance of ITER and post-ITER machines. The primary safety issue categories were discussed by S. Piet (INEL) in the context of evaluating plasma-facing component options. The design-influencing effects of beryllium-steam reactions, niobium-steam volatility, tritium implantation, and coolant ingress/plasma shutdown disruptions were reviewed. Recommendations were made on promising directions for plasma-facing components with attractive safety characteristics for ITER and machines to follow. Increased plasma-facing component research with development of defendable worst-case disruption scenarios, increased attention to mitigating chemical reaction problems, and increased attention to plasma-facing component safety issues were recommended. ITER should allow the flexibility for maximum plasma-facing component testing capability.

Magnet Engineering I
A. M. Dawson (MIT) and M. S. Lubel (ORNL)

The magnet engineering included designs for two different machines: BPX by P. Heitzenroeder (PPPL) and R. Thome (MIT), and ITER by J. Minervini (MIT). The development for BPX involved a full engineering design as one would expect for a system anticipated to be built. Of particular long-range usefulness is the work on the beryllium-copper high-strength, high-conductivity alloys, including cyclical analyses. Any copper fusion machine can benefit from this information and data base.

The development of Incoloy 908 for the conduit material in force-flow-cooled, cable-in-conduit superconductor appears to be a major advance in utilizing the full properties of Nb_3Sn. The tests of the DPC coil reported by M. Steeves (MIT) demonstrate the successful application of this material. The analysis by J. Minervini (MIT) for a full-size poloidal system for ITER confirms its usefulness. Both presentations enhanced the view that Nb_3Sn is a fully developed superconducting material, ready for application in large-scale fusion machines.

The session also included a paper by M. Werst (University of Texas) on a single-turn toroid tested to 18.1 T using power from homopolar generators, and a paper describing the BPX TF coil system design, including interesting data on beryllium-copper Bitter plate design criteria modified from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Part III.

The paper by D. B. Montgomery (MIT) discussed the design of the toroidal and central solenoid model coils for ITER and highlighted the critical issue of life-cycle testing for the magnets, as well as for the cryogenic and power supply systems. This paper tied in well with the workshop on magnet system reliability and on codes and standard.

ICF Experiments and Reactor Designs
R. F. Bourque (GA)

Erik Storm (LLNL) discussed the proposed Nova Upgrade laser, which will have a much more compact architecture than the current Nova so that it will fit into the existing facility. He emphasized the importance of developing lower yield targets with higher implosion velocities to minimize the cost of near-term experiments.

Two papers were presented on the OFE-sponsored ICF reactor studies. The first was by W. Meier (WJSA) and described two reactor designs, one for heavy ion beams and one for KrF lasers. The heavy-ion beam design, called Osiris, uses a carbon fabric tent first wall behind which flows liquid Flibe as the blanket coolant. The Flibe "weeps" through the fabric, providing a sacrificial blowoff layer, and then condenses in a Flibe pool at the bottom. With the Flibe drained out of the blanket, the empty cloth structure can be readily lifted out of the reactor vacuum chamber, as would be required about once a year. A separate paper by M. Monsler (WJSA) described the Osiris concept in more detail. The KrF design, called Sombrero, uses a carbon first wall that is cooled by fluidized Li_2O flowing behind it. An 0.5-Torr xenon gas in the chamber absorbs target X rays and debris, preventing wall vaporization. A separate paper by I. Sviatoslavsky (University of Wisconsin) described this concept in detail.

L. Waganer (MDAC) described the reactor design for a KrF laser called Prometheus. This design uses a porous ceramic structure that bleeds liquid lead to the surface, also providing a sacrificial blowoff layer. The lead vapor recondenses on the same surface, and heat is transferred to flowing lead cooling channels just behind. The blanket uses Li_2O granules that are cooled with medium-pressure helium.

R. F. Bourque (GA) described a rotating reactor concept called LIFE that uses liquid coolants like Flibe or lithium-lead. The concept, suitable for one- or two-sided illumination, has no solid first wall. There may be baffles several centimetres below the surface, however, to induce the surface turbulence needed to prevent excess vapor pressure in the chamber. With Flibe, the concept is very low activation and has low rotational energy.

The final paper of the session, by M. Tobin (LLNL), described plans for the Nova Upgrade experiment area. A low-activation aluminum 5083 alloy chamber will be surrounded by a low-activation hydrocarbon shield. With much lower target yields compared with the LMF, ~20 MJ, the chamber and final optics radii can be considerably smaller.

Fusion Machine Design
R. L. Miller (LANL) and C. Baker (ORNL)

S. A. Fairfax (MIT) described the U.S. Alcator C-Mod machine, construction of which has just been completed. This machine features flexible shaping of a noncircular, diverted plasma operated at high magnetic field (9 T) with strong ICRF heating. After initial systems integration tests are completed, operation with cryogenically cooled coils at up to 6 T will begin. A. Navarro (CIEMAT) described the Spanish stellarator TJ-II, a nonplanar-axis heliac device currently under construction and expected to be operational in 1994. This device will have a major toroidal radius of 1.5 m and operate at 1 T with ECH. E. D. Perry (PPPL) summarized preparations for D-T experiments in the TFTR beginning in 1993. Upgrades of existing hardware are under way and planned, leading to a program of 600 to 1200 D-T shots yielding 2×10^{21} neutrons in a 2-yr campaign prior to shutdown and decommissioning of that machine. A. Oikawa (JAERI) described the Japanese conceptual design of the SSTR. This 1000-MW(electric) commercial reactor would operate with a peak toroidal field at the inboard legs of the TF coils of 16.5 T. The plasma aspect ratio is 4.0, and the major toroidal plasma radius is 7.0 m. The plasma current is 12 MA, of which 9 MA is contributed by the bootstrap effect. The tritium breeding material is Li_2O in a water-cooled blanket constructed of ferritic steel with a beryllium neutron multiplier. J. Warren (Ebasco) described the results of detailed design calculations for the Inconel 718 vacuum vessel of the proposed BPX. Also, S. L. Liew (PPPL) described modeling codes and results related to the neutron and gamma radiation environment of the BPX device. With D-T operation, fully remote maintenance would be required. T. J. Herrick (ORNL) compared the implications of two alternate ITER design options (i.e., that of the CDA and the HARD) in terms of maintenance constraints, time lines, and other considerations. A mixed picture emerged, not significantly favoring either approach.

Safety and Environment
J. S. Herring (INEL) and R. S. Savercool (GA)

This session contained a mixture of theoretical and practical papers on methods for ensuring that fusion achieves its potential for safe, clean operation. S. Piet (INEL) opened the session with an invited paper on the challenges and opportunities for designing safety into ITER. He identified several areas where the current fusion program does not fully address ITER safety and environmental issues and where more work will be necessary: finding an attractive, safe divertor; developing decommissioning and decontamination practices; and demonstrating that ITER can be maintained with tolerable personnel doses. On the other hand, the CDA has already made significant progress toward developing attractive blanket options, ensuring passive safety in the blanket through natural-convection cooling during a loss-of-flow accident, and the development of low-activation materials. Piet listed four challenges in plasma engineering that will be important in the licensing of ITER: burn control of the thermally unstable plasma; a passive means of plasma shutdown; disruptions caused by coolant ingress accidents and disruption-plasma-facing component feedback; and increased burn fraction to reduce the tritium inventory. Tritium inventory, activation hazards, chemical reactions, and coolant disturbances are the challenges in divertor design. The safety-related design requirements need to be optimized because some are now too strict and others are too lax.

G. Pearson (PPPL) described the stack and area tritium monitoring systems for TFTR. Although the total on-site tritium inventory is limited to 50 kCi, the on-site consequences of a release could be severe. Area monitors provide the primary line of defense against an accidental tritium release by constantly sampling the air in various rooms at the TFTR test site. Should an area monitor go into an alarm state, automatic corrective action, such as the closing of dampers and ventilation systems, is taken immediately.

T. Fuller (MIT) reviewed the safety precautions being taken at the MIT Alcator C-Mod experiments to protect against radiation and more conventional hazards. C-Mod is expected to produce 5×10^{15} n/s for 1-s pulses at 20-min intervals. While C-Mod will not use tritium as a fuel, minor amounts of tritium will be produced through D-D reactions. Both the neutrons and the tritium will produce activated and contaminated components that must be safely handled. Some of the more significant conventional hazards are the high-power electrical systems, cryogenics, rf and microwave power, and confined spaces with a risk of suffocation. Both Pearson and Fuller presented useful perspectives for operating an experimental fusion reactor safely in the real world.

The next three papers described the safety and environmental results from design studies using the D-^{3}He fuel cycle. H. Khater (University of Wisconsin) presented results for the Apollo-L2 reactor. Primary candidate alloy, Type 316 stainless steel, and Tenelon were investigated as shield materials. At the end of reactor lifetime, only the Tenelon structure could be disposed of as 10CFR61 Class A waste. The Tenelon structure results in the highest off-site dose in a loss-of-coolant accident condition because of its ^{54}Mn and ^{56}Mn content, but the low temperature of the structure and low stored energy in the water coolant resulted in a small fractional release of the radioactive inventory. Thus, the University of Wisconsin believes that the reactor is "inherently safe," according to the Senior Committee on Environmental, Safety, and Economic Aspects of Magnetic Fusion Energy (ESECOM) definition.

S. Herring (INEL) presented the safety analysis for the ARIES-III design that uses organic coolant. Two areas of concern were activation of the tungsten on the divertor and the radioactive inventory in the shield. Isotopically tailored tungsten will result in off-site doses of only ~60 rem, even if the entire divertor inventory is released. Release of 0.5% of the radioactive inventory in the steel shield would result in lethal doses at the site boundary. Since the stored chemical energy in the organic coolant is ~6000 GJ, and since that energy could cause the release of >0.5% of the shield inventory, the design is not inherently safe, using the ESECOM definition. However, the design is clearly passively safe because no conceivable accident would cause off-site fatalities if the large-scale geometry of the reactor remains intact.

S.-K. Ho (UCB) compared the activation of ferritic and ceramic structural materials with D-T and D-^{3}He fuel cycles using the ARIES reactors. He found that the selection of materials could have a significant impact on the safety of the reactors, even if the fuel is D-^{3}He.

Finally, J. D. Lee (LLNL) presented his latest results on the possible use of an IFE reactor to transmute fission reactor wastes, particularly ^{99}Tc. He found that an IFE reactor could transmute ^{99}Tc with a support ratio of ~100 to 1 (fission power to fusion power) and that the activation products resulting from the transmutation of ^{99}Tc represent a risk three to four orders of magnitude lower than that of the ^{99}Tc itself.

Fusion Codes and Standards Workshop
P. Heitzenroeder (PPPL) and R. Thome (MIT)

This workshop was held to acquaint fusion engineers with the development process of both IEEE and ASME codes and standards. Presentations by M. E. Sheehan (ASME) and J. Foster and J. T. Bauer (IEEE) focused on the process by which codes and standards are developed. The process for developing new standards is initiated with a formal request to the society that identifies a need. The request is assigned to a committee charged with developing a draft standard and following it through a detailed approval process.

L. Sommers (LLNL) followed with a presentation titled "ITER Design Criteria." ITER will utilize superconducting magnets with stainless steel coil cases. The design criteria are modeled in part after the ASME Boiler and Pressure Vessel Code but adopt a fracture mechanics approach to fatigue.

P. Titus (MIT/Stone and Webster Engineering) presented "A Proposal for Standards for Magnet Structural Design," which reviewed a draft of a standard prepared by a task force led by H. Becker (MIT). The task force recently submitted this draft to ASME for comments. This proposed standard also is modeled in part after the ASME Boiler and Pressure Vessel Code. The need for the fusion engineering community to include structural testing and analysis of magnets to correlate design predictions is emphasized; fracture mechanics also is a part of this draft.

The last presentation of the session, titled "The Use of Statistical and Uncertainty Analyses in Designing the BPX Central Solenoid," was made by F. McClintock (MIT). This technique is used to simulate the effects of indeterminancy of input information on design limits. Based on the experience of the BPX solenoid analysis, the authors conclude that employing this method helps to focus attention on critical areas, makes for a more realistic design analysis, and should be a routine part of the analysis and be useful in further evolution of any code.

ICF Driver and Target Technology
D. L. Cook (SNLA) and J. M. Soures (URLLE)

In surveying drivers for ICF, R. Bangerter [Lawrence Berkeley Laboratory (LBL)] stated that engineering design and analysis is urgently needed to guide the development of IFE. Inertial fusion energy is a new program within energy research at DOE and currently supports the heavy-ion driver research program. Although discussion of target design details is still classified, Bangerter said that single- or two-sided indirect-drive targets are promising, that the risk of indirect-drive targets is lower than direct-drive targets, and that M. Tabak (LLNL) and C. Choi (Purdue University) have recently developed an advanced design that improves the ratio of energy in the fuel to driver energy by a factor of 2. Bangerter discussed the basic requirements for an energy-producing system, emphasizing that all of them must be met simultaneously, and summarized the status of heavy-ion and excimer laser drivers.

C. von Rosenberg (Textron) discussed KrF amplifier optimization for IFE systems. He concluded that 50 to 100 kJ is the optimum level for amplifiers, based on intrinsic efficiency, amplified spontaneous emission, gas recirculation power, and the amplifier geometric fill factor.

D. Harris (LANL) presented an improved KrF laser design for the LMF, which is more compact and lower in cost than designs from 2 yr ago. The design is based on a 400-kJ amplifier module and a mirror "roundhouse" for angular multiplexing.

G. Linford (TRW) presented a multimegajoule KrF laser driver design for an IFE reactor in 2050. The design is based on Raman accumulators (60 beams, 100 kJ each), computer-adjusted alignment for each mirror made possible by massively parallel computers, and pyramidally apodized beams for 1% uniformity on the direct-drive target. The basic amplifier module in this system was 4 kJ.

J. Woodworth (LLNL) discussed the final optical element in a laser-driven IFE reactor. An experimental study of 350-nm light absorption in neutron-irradiated fused silica showed that thermal annealing removes much of the damage. The solution to a set of rate equations describing damage creation and annealing showed that 300°C operation of fused silica provides 99% transmission even though the optical wedge gets 0.1 Mrad/s at 40 m from the reactor center.

D. Drake (KMS Fusion) presented view factor calculations of beam misalignment, beam timing jitter, and target injection inaccuracies on irradiation symmetry in an IFE reactor. The results were given as output from a Silicon Graphics workstation and provided an impressive intuitive feeling for the effects of imperfect illumination.

D. Beller (Air Force Institute of Technology) discussed a new three-phase 7-yr U.S. Department of Defense/DOE program on antiproton-boosted ICF. The concept is to spark D-T fuel with antimatter for boosting the yield. Applications are space propulsion and energy production. The idea is to use a radio-frequency quadrupole accumulator to slow and trap antiprotons from CERN (now making 100 million antiprotons per day) and apply them in an experiment on the Shiva-Star facility at Phillips Laboratory in an attempt to measure boosting.

Heating and Current Drive
D. W. Swain (ORNL) and D. W. Sedgley (Grumman)

The invited paper by T. Wade (JET) described the injection of up to 22 MW of ICRH power into JET. Limitations on the injected power have been identified; for high values of plasma loading, the limitation was caused by breakdown in the generator caused by high generator voltages induced by momentary breakdowns near the antenna. The voltage capabilities of the generator components are being improved to counteract this effect. The JET ICRH system now has the capability to run with the antennas at ±90-deg phasing as well as the earlier values of 0 and 180 deg.

W. P. Cary (GA) described the installation and testing of the 110-GHz system on DIII-D. The system uses four Varian gyrotrons. A test gyrotron has produced over 500 kW for 5 s into a dummy load. Low-power tests of the ECH transmission and launching system indicate that the transmission efficiency to the plasma from the gyrotron should be >90%.

A study of the changes in the ITER heating and current drive system that would be necessary for a HARD was described by W. Nevins (LLNL). For a higher aspect ratio, the bootstrap fraction is higher, and the physics requirements are for a more peaked current profile. Since the lower hybrid system cannot drive current in the center of the plasma, it is not possible to achieve the peaked current profile required with a lower hybrid system. For an aspect ratio of 4.0, the system would consist of neutral beams only for current drive, with no lower hybrid system. The total beam power is 110 to 120 MW, approximately equal to the total of beam and lower hybrid power for the reference design. Electron cyclotron current drive is proposed for profile control.

K. Yoshikawa (Kyoto University) described a plasma neutralizer for neutral beams that would be suitable for ITER. For ITER, 75 MW of neutral beams at 1.3 MeV will require 163 MW of beam power if a gas neutralizing cell is used; this number drops to 115 MW if a plasma neutralizer is employed. A neutralizer that employs a magnetron-type discharge to generate the plasma with density of 0.8×10^{18} m^{-3} was described that is near to meeting the requirements for the ITER beams.

The neutral beam heating results on TFTR during 1990 were described by T. Stevenson (PPPL). Continued improvements in the beam system resulted in an overall beam availability of ~80%, for a total of 3000 machine pulses. They achieved up to 24 MW for 2-s pulses for a total energy delivery of 48 MJ/shot. Up to 33 MW was delivered for 0.8 s into TFTR supershots, resulting in 5×10^{16} neutrons.

A modified neutral beam system that injected neon beams into TFTR was described by J. H. Kamperschroer (PPPL). Up to 8.2 A of Ne^0 at an energy of 60 kV has been injected into the plasma for impurity transport studies. They plan to inject 120-kV Xe^0 after the installation of a higher current bending magnet. A significant limitation on the injection of high-Z species is the occurrence of a small but nonnegligible number of doubly charged ions in the acceleration region. These ions then gain twice the acceleration voltage and then strike the ion beam dump with significant amounts of power; for example, for 8 A of 120-kV xenon, ~0.5 A of Xe^{++} at 240 kV will be accelerated.

Remote Handling and Structural Systems
F. C. Davis (ORNL) and T. G. Brown (PPPL)

This session provided overviews of the remote handling needs of planned tritium-fueled fusion machines. Presentations were given on remote maintenance for BPX by F. C. Davis (ORNL), T. G. Brown (PPPL), T. W. Burgess (ORNL), and D. P. Kuban (TeleRobotics). ITER remote maintenance was discussed by D. C. Lousteau, and Next European Torus (NET) remote handling systems by C. Holloway (NET). Generic remote maintenance issues were covered by T. Haines (Spar Aerospace). The importance of including remote handling at the conception of the project was emphasized. JET, and to a lesser extent TFTR, have limited remote handling capability, which restricts the ability to recover from failures. Each of the future machines (BPX, ITER, and NET) have as a project requirement the ability to recover from component failures; the degree of commitment to the requirement varies. BPX has the requirement to recover from the failure of any operating component, including a coil failure. BPX is designed in a modular fashion; any module can be remotely replaced. Changout of the most difficult module to be remotely replaced, a TF coil, can be accomplished in ~2 yr.

ITER has the same basic requirement of remote recovery from a component failure, but there is concern regarding the feasibility of remotely replacing one of its TF coils, particularly with today's technology. A TF coil is so large and the replacement task so demanding that the time required may be unrealistic. NET was not as specific as to their remote maintenance requirements. For each of the future machines, the need for comprehensive R&D programs and mockup demonstrations was emphasized. Standards must be developed for components that are to be remotely handled, equipment to perform the remote maintenance tasks must be developed, and critical remote operations must be demonstrated in full-scale mockups. Remote maintenance must be a requirement from the start of the design process and must retain the same degree of importance as all other requirements for each and every component to be remotely maintained.

Fusion Power Reactors II/Blanket and Shield Engineering
J. Doggett (LLNL) and C. P. C. Wong (GA)

G. A. Emmert (University of Wisconsin) reported continued refinement of the Apollo-L3 D-^{3}He tokamak design. The revisions to the design basically consist of increases in size. The thrust of the design continues to be minimization of activation products and off-site accident radiation doses by use of the D-^{3}He fuel cycle, while retaining use of "conventional" materials.

M. E. Sawan (University of Wisconsin) presented first-wall and shield designs for the ARIES-III D-^{3}He tokamak. An organic-cooled ARIES-III is possible because of the ability with the D-^{3}He fuel cycle to keep radiation levels and temperatures low enough to reduce coolant decomposition to acceptable levels. The use of organic coolant aims at reducing mechanical loads through its ability to remove heat adequately at low pressure.

R. L. Miller (LANL) reported on a systems analysis model that has been developed to compare pulsed tokamaks with steady-state machines. The model appears to need refining to correctly model parameters such as bootstrap current. Pulsed or hybrid-pulsed operation seems to increase the COE beyond that of a steady-state device.

An assessment of the driver blanket for ITER was presented by A. R. Raffray (UCLA). The driver blanket and the nonbreeding shield designs will have similar levels of design complexity. Purchasing tritium for ITER will generate safety problems in transportation. His key concern with a solid breeder blanket is whether the tritium would come out from the breeder materials fast enough. He advocated reconsideration of a helium-cooled driver blanket for ITER.

The magnet shielding effectiveness of the U.S., USSR, Japanese, and European blankets for ITER were compared by L. A. El-Guebaly (University of Wisconsin). All the designs will need to be modified to meet the magnet protection criteria. The USSR, Japanese, and European blankets will need to be thicker.

The ITER blanket and shield studies for the HARD option were presented by Y. Gohar (Argonne National Laboratory). The HARD makes more efficient use of fusion neutrons than the baseline $A = 2.8$ design. By allowing back heat flux from the blanket to the first wall, the thermal- and pressure-induced displacements and forces can be balanced. A more robust first-wall design is possible.

Magnet Reliability Workshop
F. Puhn (GA) and D. B. Montgomery (MIT)

The Magnet Reliability Workshop was a series of oral presentations by experienced magnet engineers to address the problems of attaining reliability in magnet systems. The experience of both the fusion and Superconducting Supercollider programs was well represented. The information presented ranged from detailed problem-solution descriptions to broad general advice on how to avoid problems in the future.

The recurrent general themes offered by the workshop participants could be summarized as follows:

1. The things we worry about the most during design tend to fail the least.

2. Little things tend to fail, particularly those things that get minimal attention during design.

3. We need more engineering attention on details to improve reliability (and this will cost more money than we are used to spending).

4. There are specific failure modes and effects that seem to repeat themselves, and these need more attention.

In the same vein, the recurrent specific failure modes offered in one form or another by the participants could be summarized as follows:

1. faults in electrical insulation caused by contamination or other unknown defects
2. friction in clamped interfaces being too high or too low, or intermittently disappearing, causing excessive forces, ratcheting deflection, or slippage
3. structural stiffness being lower than anticipated in complex assemblies (perhaps by a factor of 2 or more)
4. leaks (coolant, vacuum) that propagate into other more severe failures
5. loads not being completely defined, particularly secondary loads due to fringe fields, friction, excessive deflection, etc.
6. failures that occur or propagate because of inadequate information on the operational status of the magnet.

A general consensus of the attendees was that a design handbook for magnet engineers is needed. Advice, historical facts, and rules-of-thumb such as those presented in these talks could all be included in such a handbook. The focus on improving magnet reliability stems from a need of large, future fusion projects such as ITER. The subject is becoming of primary importance to major programs and must receive more attention in the future.

POSTER SESSIONS

There were six poster sessions, one every morning and afternoon, each divided into subsessions by topic. The large number of papers in these sessions generally precludes discussion of each paper. Some trends and highlights are noted, however, from each session.

Structural Systems
J. Kim (GA)

J. Citrolo (PPPL) presented the overall coil support design features for BPX's 9-T TF coils. A wedged inner-leg support concept is incorporated, with six three-coil modules to facilitate remote replacement. B. Nelson (ORNL) discussed the effect of the ITER HARD design on the containment structures. The magnet overturning forces are doubled over the baseline design but can be accommodated by changes in the coil structures.

F. Dahlgreen (PPPL) described a computer simulation of structural response to a seismic load on BPX for known seismic activities around the Princeton area. Seismically induced stresses are not of much concern.

E. Reis et al. (GA) reported measurement of vertical motions of the DIII-D vacuum vessel during disruptions over the past 4 yr and the accompanying theoretical modeling. A maximum displacement of 0.062 in. during a 1.7-MA plasma disruption was observed. Their theoretical model predicted the measured vessel motions very well.

Plasma Heating and Current Drive Systems
J. H. Kamperschroer (PPPL)

Positive ion neutral beam systems on the large tokamaks (JET, TFTR, and DIII-D) have matured, as evidenced by their high reliability (>80%) and by the exotic gas species sometimes used. In addition to the usual hydrogen isotopes of protium and deuterium, helium (^{3}He and ^{4}He), neon, and argon beams have successfully been operated; plans to use even heavier gases are being formulated. Both JET and TFTR plan to introduce tritium into their beamlines within the next few years, and the design of the TFTR beam gas injection system was presented. Power measurements on the DIII-D beamlines have been refined, reducing the errors inherent in the injected power measurement.

Future neutral beam systems will require higher energies and will, by necessity, have negative ion sources. Several plans for such negative-ion-based neutral beam systems were given by the Japanese. The beam group at JAERI is carrying out R&D on a system capable of extracting 10 A at up to 1.3 MeV for long pulses and is planning a system to inject 10 MW at 500 keV for 10 s into JT-60U. In parallel, the National Institute for Fusion Science is constructing a test facility to develop technology capable of injecting 20 MW of neutral power from 125-keV hydrogen or 250-keV deuterium. Most of these negative ion systems rely on doping the hydrogen or deuterium plasma with cesium to enhance the production of negative ions. An alternative technique of using a coating of barium on a substrate to enhance the negative ion production was presented by LBL.

Design studies were presented for the ARIES-III D-^{3}He reactor, indicating that neutral beams with energies up to 6 MeV may be required. Designers of the ITER HARD option are considering current drive via a combination of bootstrap current, 1.3-MeV neutral beams, and lower hybrid waves.

Designs of many wave heating systems were presented. Ion cyclotron heating systems are planned for C-Mod, JT-60U, and the Large Helical Device (LHD). Eight megawatts at 80 MHz is planned for C-Mod; on JT-60U, two new launchers replaced one on the old JT-60 configuration; and ICRF is planned for 3-MW continuous-wave use on the LHD. A 20-MW ICRF system also was planned as a heating source for BPX. An upgrade of the TFTR ICRF system to a total of four antennas and a total power of 12.5 MW was described.

Several wave-based techniques are being pursued for current drive. Lower hybrid current drive is planned on JT-60U and PBX-M, while the Frascati Tokamak Upgrade (FTU) will use 9-MW lower hybrid waves at 8 GHz to heat the plasma. Innovative current drive techniques of a 2-MW fast-wave phased array on DIII-D and 5-GW pulses at 140 GHz from an FEL on MTX are planned for the near future.

Fusion meetings for approximately the next 5 yr should be very exciting as the effort expended in the many varied heating and current drive techniques described at the symposium begin to bear fruit and the results of experiments begin to be published.

Magnet Engineering
A. L. Langhorn (GA)

This poster session was well attended with 26 presentations by contributors from the United States, People's Republic of China (PRC), Japan, and Spain.

The session was dominated by presentations regarding BPX. Eleven papers charted the progress of work in support of coil design, analysis, and development. Global and detailed local stress analysis has been performed on the central solenoid and the TF coils. Several papers described selection, characterization, and testing of the conductor and insulation. A special beryllium-copper alloy has been developed that will meet the stress, electrical, and fabrication requirements of BPX. Since the coils in this device operate from 77 to 300 K, additional stress analysis and testing has been performed and was reported here.

Other papers in the session dealing with coil analysis, design, and fabrication included contributions from Alcator C-Mod and ARIES-III (MIT), Experimental Hybrid Reactor (EHR) (PRC), and LHD (Japan). A trend toward cryogenically cooled coils, both resistive and superconducting, was noted. The goal is one of cost savings: Subcooling of the coils on a pulsed machine reduces the initial resistance and allows longer pulses before the coils reach temperatures where insulation breakdown occurs; the cost is added complexity in coil support structure and thermal insulation. Superconducting coils require better thermal insulation and cost more to fabricate. However, they have the advantage of continuous operation and lower coil power cost. One extremely valuable contribution was a paper by Wu et al. (Westinghouse), describing a high-T_c, low-loss power lead for high-current superconducting magnets. The design presented would save perhaps 40% of the power costs of the cryosystem and reduce the capital cost of the equipment.

In the area of device operations, Marsala et al. (PPPL) reported on the use of an electrical test setup/procedure to locate and characterize multiple electrical leak paths and turn-to-turn shorts in the TF coil set. This problem is universal in operating machines, and the approach described represents a useful method of monitoring coil integrity.

Vacuum, Divertor, and First Wall

This session consisted of 16 high-quality papers reflecting the active interest in the vacuum vessel and plasma-facing components as critical features of current fusion machines and even more critical aspects of future fusion reactors. Two papers described useful results from DIII-D. J. P. Smith (GA) reported on the evolving design of the electrical insulation system for the biasable advanced divertor electrode. Standoff of up to 5 kV in a plasma environment is a significant challenge, but successful design solutions are emerging. T. Hodapp (GA) described the boronization system used in recent successful experiments to reduce plasma impurities. The system has numerous safety features to safely handle the toxic and pyrophoric diborane gas used to deposit a 100-nm boron film on the plasma-facing surfaces by plasma-enhanced chemical vapor deposition.

Testing programs for plasma-facing components and materials are yielding useful results. G. Sannazzaro (JET) described low-cycle fatigue tests of the Inconel 600 JET vacuum vessel materials. Detailed analysis applying these results to lifetime estimates of the JET vacuum vessel show ample margin for continued and extended operation. M. Araki (JAERI) reported on thermal cycle tests of ITER/FER first-wall and divertor components in the extensive JAERI plasma-facing component development program. Experimental simulation of plasma disruptions on graphite by M. A. Bourham [North Carolina State University (NCSU)] indicate that plasma vapor shielding can significantly reduce material ablation. At very high heat fluxes, up to 96% of the disruption energy can be radiated. J. F. Crawford (University of New Mexico) described a coaxial plasma gun system recently built to study disruption effects on various material samples.

Three papers described various aspects of the European plasma-facing component development programs. E. Di Pietro (ENEA) presented the status of the Monobloc divertor concept development. P. Deschamps (CEN) reported on testing of high-heat-flux components for NET and ITER divertors, including testing in Tore Supra. G. P. Adorno (Ansaldo) described development of manufacturing processes for fabrication of NET/ITER first-wall panels using a new brazing process.

Plasma-facing component designs for future experiments and reactors were also presented. F. R. Williams (MDAC) summarized design and analysis of the BPX inboard limiter. Carbon-carbon tiles mounted in metal frames withstand the severe thermal and electromagnetic loads. M. Hasan (UCLA) presented the design of the ARIES-III divertor. A tungsten-rhenium alloy structure is needed to allow coolant temperature adequate for power conversion with the high heat fluxes found in this D-^{3}He fueled advanced reactor. The organic coolant allows high coolant temperature at low pressure. H.-W. Shi (Southwestern Institute of Physics) submitted a paper on the plasma-facing component design of the Chinese Tokamak Engineering Test Breeder conceptual design study.

Analytical studies offered insight into plasma-facing component behavior. M. Shibui (Toshiba) showed analysis of edge stresses for bonded armor tiles on divertor plates. He proposed a conical bonded interface that minimizes stress concentration at the edges of bond interfaces. K. A. Niemer (NCSU) has done an analysis of runaway electron damage in the presence of magnetic fields. The magnetic field will reduce the depth of penetration of the electron beam, but this will cause more extensive surface damage. Magnetic effects must be included in analysis of runaway electrons. M. Hasan (UCLA) similarly showed the importance of including the effects of nonuniform circumferential heat flux and local magnetohydrodynamic (MHD) effects in the analysis of liquid-metal-cooled divertor tubes. Poloidal edge plasma currents have been recognized recently as major contributors to the forces on plasma-facing components and vacuum vessels during tokamak disruptions. J. G. Murray (consultant) pointed out that both good clean vacuum systems and the edge current conditions affect this poloidal current. He suggested that control of the edge cross currents can provide an effective and efficient mechanism to control the poloidal currents.

Electrical Power Systems
D. B. Remsen (GA)

The papers in this session showed that there is much work being done in the area of electrical power systems. Several refinements are being incorporated into existing or upgraded power systems, and new power systems design and fabrication work was reported. In general, the session indicated that a serious effort is being made by all contributors to emphasize reliability of operation in power systems that support fusion experiments. The expense of each shot on a large machine dictates that all systems operate with high reliability to maximize the data obtained from that shot.

Several papers reported work with model or prototype power systems being tested before building full-scale systems to verify and test new approaches on a modest scale prior to committing the large resources needed for the full system. Increased interaction between laboratories was noted, to incorporate the maximum experience possible into the design of new systems. This, in turn, maximizes the probability of success and minimizes reinvention and duplication of effort. These trends are especially positive in the present difficult budget situation.

Remote Handling Systems
D. C. Lousteau (ORNL)

Remote handling efforts on pipe cutting and neutral ion beam source replacement at JET were reported by S. Mills and A. Rolfe (JET). Both papers were based on the use of mockups to gain actual hardware experience and showed the resulting first generation of tools. Cutting of both Inconel and stainless steel pipes without the use of cutting fluids was successfully demonstrated, and ideas were presented that could lead to improved performance. The neutral beam mockup demonstrated the need to establish the design and arrangements of the interfaces in order to be compatible with remote handling requirements. The experience of this work resulted in a number of modifications to the remote handling equipment, the beam source design itself, and ultimately, the successful remote replacement of a beam source.

Studies on proposed tokamak reactors have identified the need for several essential developments in remote handling technologies, including tools for handling large in-vessel components, remotely maintainable large vacuum gate valves, support and alignment systems for assembly of the reactor, and locking systems for in-vessel components. A unique modification to a vacuum gate valve was shown by J. Stringer (Wardrop) that allows the gate and actuating mechanism to be removed for maintenance while preserving vacuum in the line. Two auxiliary rectangular isolation gate valves are mounted in the main valve between the bonnet and valve body. When the main valve gate is in the open position, it can be isolated and separated from the valve body by closing these two auxiliary valves.

Disruptions in reactors with large plasma currents result in large electromagnetic loads on in-vessel components, specifically divertors and blankets/shields. Coupled with the problem of gaining access to affected components, this consideration results in need to develop special-purpose support systems and fasteners. The Japanese, in papers by H. Fukushima (Toshiba) and S. Nishio (JAERI), are proposing fasteners utilizing hydraulic actuators for this purpose, both in ITER and the Fusion Experimental Reactor (FER), and have conducted tests and evaluations of three types: a telescoping hydraulic jack, a hydraulic cotter, and a flexible hydraulic tube. The test successfully demonstrated the ability of all three to provide the required load for the required number of cycles. A separate paper by K. Shibanuma (JAERI) showed the one-fifth-scale mockups used to demonstrate the handling of divertors and blankets. Using these mockups with specially constructed overhead handling equipment and a rail-with-vehicle system, JAERI has successfully demonstrated the removal of these in-vessel components.

Remote maintenance systems, including those for present-day and proposed fusion machines, rely on a "human-in-the-loop" control philosophy utilizing servomanipulators mounted on positioning devices. Engineering problems being encountered in the development of these systems, and discussed during the session, include control of deflections/vibrations on the positioning devices and operator control of the manipulator. L. Galbaiti (JET) reported JET is applying a Mascot 4 manipulator mounted on an articulated boom for its in-vessel maintenance needs. Efforts to increase the dexterity of the manipulator by improving its stiffness, sensitivity, damping, maximum operating speed, and time response were discussed along with the integration of computer-aided controls and a viewing system. A separate paper by M. Cotsaftis (UR/CENFAR) discussed a proposal for a mathematical technique and a proportional derivative-based joint control system that can generate joint torques to control deformation modes of the articulated boom system under payload-induced shock loading. The proposed system utilizes an accelerator sensor at the end of the boom to provide real-time response and is reported to be very insensitive within ranges to variations of the specified payload.

Health, Safety, and the Environment
S. K. Ho (UCB)

The seven papers presented at this poster session addressed issues relating to current experimental tokamaks, near-term test reactors, and long-term development. These progressive steps indicate an emerging safety and environmental concern for fusion.

R. Ross (JET) reported that the JET program has taken up measures to ensure the highly toxic beryllium dust from the beryllium first wall is under control. Beryllium control areas with ventilation and changing room facilities have been established. Diagnostics and work related with external features of the machine are performed using flexible gloveboxes to prevent contamination of the building. Air samples and wipe samples are used to track the beryllium concentration for both airborne and surface contamination. As a result, the beryllium dust is adequately controlled.

The DIII-D experiment can generate $\sim 6 \times 10^{16}$ D-D neutrons per shot. A radiation shield, described by P. Taylor (GA), was built to minimize the site boundary dose level. The measured site neutron and gamma dose levels, which agree with design calculations, are monitored and stored in a data base. The dose is well below the DOE administrative level of 20 mrem/yr.

Risk analysis has been performed for the ITER and NET designs and was reported by A. Bosini (ENEA) and G. Cambi (University of Bologna). Systematic functional analyses with the event-tree technique have been performed for the accident scenarios of the ITER divertor and first wall. The first wall and divertor have similar accident risks. Active isolation valves and passive filters and vents are used to safeguard against LOCAs and loss-of-vacuum accidents. The absolute radioactivity inventory exceeds the quantity that could cause an external dose >0.1 Sv, the limit for public protection. However, the risk, calculated by the product of the failure frequency and the dose value, is four orders of magnitude below the public safety criterion.

A similar approach also is applied to a loss-of-isolation-vacuum accident analysis of the NET magnet design, as reported by R. Caporali (ENEA). A water-ingress accident can trigger overheating of the coils, formation of electrical arcs, and spilling of helium into the cryostat. Relief valves and tanks should be installed to prevent any overpressurization of the coil due to the heated helium.

Low-activation structural materials such as reduced-activation ferritic (RAF) steel, vanadium, and SiC composite were investigated by E. T. Cheng (TSI Research) for future fusion power reactors based on the criteria of afterheat, accident dose, maintenance, waste disposal, and recycling. Silicon carbide has no intrinsic activation problem; and the removal of the small fractions of impurity elements and isotopic tailoring can greatly reduce the activation inventories for the RAF and vanadium. Impurity removal and control for both the structural materials and breeders are very critical for hands-on maintenance and waste disposal. Work has started investigating the technological feasibility of controlling impurities to <1-ppm levels during the fabrication process.

A neutronics analysis was presented by H. Khater (University of Wisconsin) to model the mixed D-T and D-D neutron fluxes for a D-^{3}He reactor. Neutronics calculations incorporating the two neutron sources simultaneously, rather than adding the individual contributions of the two sources, are essential for determining the burnup effect, especially for large fluence and reaction cross sections. The omission of the burnup effect accounts for a 25% overestimation of the prompt dose in a typical 30-full-power-year D-^{3}He reactor.

Plasma Fueling and Particle Control

There was strong emphasis in this session on design, development, and testing of light gas gun pellet injectors. Papers on pneumatic guns from the United States, the European community, and Japan indicated this technology is maturing well, and the prospects are excellent for reliable delivery of D-T ice pellets at speeds up to 3 km/s. S. L. Milora (ORNL) showed the design for the tritium pellet injector now being constructed at ORNL for installation on TFTR. Four barrels will provide capability for injection of 3.5- to 5.5-mm-diam pellets at speeds to 1.5 km/s with single-stage drivers or to 3 km/s with a two-stage driver. A. Frattollilo (ENEA) showed the results from design, construction, and testing of a two-stage gas gun injector for deuterium pellets into the FTU. Pellet speeds of up to 3.2 km/s have been measured. Similar speeds have been achieved with a two-stage gas gun designed and built for Heliotron-E, as reported by M. Kanno (Kobe Steel). Designs for repetitive operation are now in progress. A. Shigenaka (Hitachi) reported on development of a single-stage gas gun capable of accelerating hydrogen ice pellets to 1.6 km/s in steady-state operation at 1 Hz, with bursts of 10-s duration at 5 Hz. Several techniques for improving the performance of two-stage pneumatic guns were described by A. Reggiori (CNR). Smoother barrels of low-conductivity material and use of a resonant chamber at the barrel entry yield significant improvement in velocity. Speeds of up to 6 km/s have been measured with plastic pellets.

Several other hydrogen ice pellet acceleration techniques were presented that are in earlier stages of development. C. C. Tsai (ORNL) discussed results from tests on a proof-of-principle electron-beam pellet accelerator that reached speeds of 0.4 km/s. Several design improvements, including a high-temperature cathode and a casing around the pellet, were proposed for a new, higher velocity gun. Y. Oda (Mitsubishi) described progress on development of electromagnetic rail guns for hydrogen ice injection. Tests with plastic pellets are now under way, with plans and designs for a system capable of ice pellet injection at speeds of 2 to 5 km/s.

The design of pellet fueling systems for the tokamak commercial breeder reactor was the focus of a parametric study submitted by B. Q. Deng (Southwestern Institute of Physics). The computer code written to carry out the parametric analysis includes variations of all major plasma engineering parameters. Pellet fueling was used as a tool to investigate energy and particle transport in the Heliotron-E experiments. S. Sudo (Kyoto University) reported on experiments to study the effect of auxiliary TF control, magnetic axis shift, and external field perturbations with and without pellet injection.

K. Holtrop (GA) described the addition of an electron gun to DIII-D to assist initiation of helium glow discharge for first-wall cleaning. The electron beam reduces the pressure required to initiate the glow discharge that greatly reduces the requirement for cycling of vacuum valves during discharge cleaning.

Control, Instrumentation, and Data Handling
J. C. Phillips (GA)

The 40 posters in this session covered topics ranging from broad conceptual designs for new tokamak control and data analysis systems ["Remote Experimental Site: A Command and Analysis Center for 'Big Physics' Experimentation," by T. A. Casper (LLNL) and "Fusion Reactor Control," by D. A. Plummer] to detailed descriptions of specific control and acquisition upgrades to existing machines such as the Texas Experimental Reactor (TEXT), and Alcator C-Mod.

Exciting real-time and near-real-time computing applications were presented, such as "An Advanced Plasma Control System for the DIII-D Tokamak," by J. R. Ferron (GA), and "Applying Neural Networks to Control the TFTR Neutral Beam Ion Sources," by L. G. Lagin (PPPL). J. Kalnavarns [Tokamak de Varennes (TdeV)] described the plasma control system used on the TdeV. Real-time feedback control of the plasma vertical and horizontal position, plasma current, and density is provided by a programmable high-speed controller coupled to a 386 personal computer running commercial process control software.

The integration and use of commercial software products for fusion applications was addressed by three papers: "The TdeV Timing System," by P. deVillers (TdeV); "Synchronization of Timing Systems on TFTR," by E. J. Montauge (PPPL); "Timing System for Neutral Beam Injection on the DIII-D Tokamak," by G. Bramson (GA); and "A Modular Timing System for Megawatt Gyrotrons," by D. E. Petersen (LLNL). The paper presented on the subject from TdeV presented interesting commercially available computer-automated measurement and control (CAMAC) hardware using a noise-immune fiber-optic clock.

The subject of the large volume of data acquired by large physics experiments, such as tokamaks, and its handling and reduction, was addressed by several papers: "File Storage Management for TFTR Physics Data," by C. Ludescher (PPPL); "The Use of a VAX Cluster for the DIII-D Data Acquisition System," by B. B. McHarg, Jr. (GA); and "Diagnostic Data Management on MTX," by D. N. Butler (LLNL).

Plasma Limiter, Divertor, and Heat Removal Systems
C. B. Baxi (GA)

Heat removal in high-heat-flux regions such as limiters and divertors is one of the critical issues in the design for the next generation of fusion devices such as STE and ITER. Eight papers were presented in this session.

Two papers dealt with the ITER divertor design. J. C. Wesley (GA) discussed thermal effects of divertor sweeping. He concluded that although divertor sweeping helps in reducing the peak temperatures, an increase in sweeping frequency to 1 Hz was required to provide a power handling margin of ~25% for ITER. P. L. Goranson (ORNL) discussed the design criteria for the mounting studs for the ITER divertor and concluded that development of new components and innovative configurations is necessary to satisfy the design criteria.

Three papers were devoted to upgrade of present machines. G. Brolatti (ENEA) presented a study of the toroidal limiter for the FTU machine. The toroidal limiter is required because of the additional 8 MW of power that will have to be handled during the 8-GHz rf heating phase.

G. W. Barnes presented the design of the TFTR rf limiter upgrade capable of handling 50 MW of power for 2 s. The tile material will be two-dimensional carbon-carbon composite. K. Ioki (Mitsubishi) presented the design of a water-cooled divertor base plate for JT-60U. This design can handle a heat flux of 20 MW/m^2 for 5 s. D. J. Strickler presented the plasma shape control calculations for the BPX divertor. R. A. Vesey (Rensselaer Polytechnic Institute) discussed an innovative, but probably impractical, concept of a moving belt divertor.

A. Doria (ENEA) presented an X-ray method used to test the quality of brazed joints between graphite and molybdenum. Such testing was necessary for the RFX upgrade, which used graphite walls.

Vacuum Vessel Design and Performance
D. D. Lang (LLNL)

This poster session had nine papers that discussed vacuum vessels for TFTR, JT-60U, BPX, ITER, and TJ-11. The papers covered a useful mix of existing hardware, newly fabricated and tested hardware, and analysis and design for future machines.

The TFTR paper by G. Loesser (PPPL) described an arm that measures the position of any interior first wall or other components from a reference plane set up on the horizontal axis of the vacuum vessel. This is a fairly critical issue as the heat loads depend on the proper positioning of plasma-facing components. This manually positioned arm produced good results using optical encoders and a joint design that had essentially no play. The results showed that some components were misaligned to the magnetic field by up to 10 mm.

Three papers by K. Takanabe, T. Masuzaki, and T. U. Chikaowa (Mitsubishi) discussed the vacuum vessel for the JT-60U tokamak, which was completed in spring of 1991. The basic design is a thin double-skin torus with a D-shaped cross section. The vessel had a 9.3-m outside diameter and a 3.3-m height and was fabricated out of Inconel 625. Innovative processes were used in the fabrication of the vessel to minimize distortion, including three-dimensional CAD/CAM operation and three-dimensional forming using hot sizing at 850°C, along with port cutting by abrasive jet. The thin Inconel skins provided the required toroidal resistance, and the intermediate volume provided the heating and cooling passages to bake the vessel to 300°C as well as removal of the heat from plasma shots, using a gas circulation system. The support system for the JT-60U vacuum vessel was unique in that special sliding supports, which are installed at the inner diameter and at the outer diameter on the horizontal midplane, had to be developed to handle the dead-weight loads and electromagnetic loads as well as accommodate the thermal motion.

The two papers by P. Hsueh and S. Dinkevich (Ebasco), discussed the structural analysis of the conceptual design for the BPX vacuum vessel and the analysis and testing program for the vessel segment bolted joints. The BPX design is for a thick-wall Inconel vessel. The structural analysis is an update of previous analysis, incorporating a major change in the model symmetry due to port changes, relocation of the vertical support from the midplane to the bottom of the vessel shell, and a revision of the finite element model from an 18-deg segment to a 60-deg segment. The bolted-joint design for the vessel has been analyzed and tested, and a detailed discussion of this program was reported.

The ITER double-wall vacuum vessel concept was analyzed for both stress and electromagnetic loads on ITER in-vessel components in two papers by D. Conner and D. Williamson (ORNL). The stress analysis showed that a thin, double-wall sandwich-concept vessel is a feasible alternative to the ITER baseline thick-wall design and could easily be heated and cooled. Significant electromagnetic loads were seen on the internal blanket/shield segments, and methods to reduce those loads were explored.

The complicated vacuum-vessel design for the TJ-II stellarator was described by J. Botija (CIEMAT) in the last poster of this session. This vacuum vessel is similar to the ORNL Advanced Toroidal Facility (ATF) vacuum vessel. The vacuum vessel follows the helical arrangement of the TF coils. A concave helical surface groove allows the central conductors to be closely fitted. The basic material is Type 304 LN stainless steel. A detailed, integrated assembly sequence is followed as the magnets and octants of the vacuum vessel are brought together.

Fusion Machine and Reactor Designs

This session covered a broad spectrum of papers, ranging from neutronics studies and tritium systems analysis to magnetics and magnet design to systems studies and description of machine design. E. T. Cheng (TSI Research) detailed the reduction in fusion blanket performance due to the latest ENDF/B-VI beryllium cross sections. X. Gao (IPP Hefei) described radiation damage studies for the tokamak experimental hybrid reactor that indicate higher radiation damage than for a pure fusion reactor.

M. Viola (PPPL) presented the design of a cold trap system for TFTR that will protect the tritium recovery and cleanup system from catalyst poisoning by SF_6 that may leak into the vacuum system from various insulation applications. Tritium inventory is a more serious concern for the BPX design. To avoid excessive inventory, an inventory management scheme, described by M. Ulrickson (PPPL), will be used to remove tritium from the carbon first-wall tiles by helium-oxygen glow discharge cleaning. E. Santoro (ENEA) submitted the description of the TRICHECO facility designed for insertion in a TRIGA reactor to study the release of tritium from candidate solid breeder materials.

Plasma magnetics analysis for several machines was presented. R. LaHaye (GA) described error field analysis for BPX and suggested schemes to avoid locked modes and magnetic islands that such error fields can induce. C. Kessel (PPPL) presented analysis of vertical and radial plasma position control for BPX. Control schemes and control coil locations to properly effect plasma control were described. R. H. Bulmer (LLNL) examined the impact on the poloidal magnetics system of the HARD for ITER. The HARD enjoys a lower plasma current than the reference $A = 2.8$ design, and no negative impacts on the poloidal magnetics were found. L. Liu (IPP Heifei) submitted the conceptual design of the plasma vertical control system for the experimental hybrid reactor that uses passive and active control coils to stabilize plasmas at up to elongation of 2.

Systems analysis was presented by L. Bromberg (MIT) on commercial fusion reactors with resistive coils. Physics regimes with potential for high beta improve the prospects for viable copper coil reactors. The cost savings of resistive coils cannot compensate for the high recirculating power required to power the coils, however. The benefits of apparently intangible features such as potential higher reliability and maintainability must be invoked to justify such designs. G. Emmert (University of Wisconsin) presented a detailed comparison of startup of the ARIES-III D-^{3}He reactor using a D-T "match" rather than pure D-^{3}He. The D-T startup minimizes auxiliary power requirements but imposes constraints due to neutron effects.

Several magnet design studies were presented. L. Bromberg (MIT) studied the possibilities of steady-state operation of the BPX magnets by use of active liquid nitrogen cooling. Numerous design changes to the coil details will allow BPX to achieve steady-state operation. R. D. Pillsbury (MIT) presented detailed analysis of the impact of startup and discharge cleaning requirements on the PF system for BPX. J. Schultz (MIT) described the PF system for the HARD version of ITER. The lower plasma current and larger volt-second capability of HARD allow a fourfold increase in pulse length and thus neutron fluence to test modules.

The papers describing various machine designs ranged from descriptions of existing experiments to proposals for future advanced reactors. Y. Sato (ETL) described the TPS-2M compact RFP machine, completed in summer 1991, built to study edge plasma control, current profile control, and stabilization with a thin conducting shell. The design evolution of the BPX configuration was presented by G. Carguilia (PPPL). Four distinct design configurations for the TF coil structural design were studied over the 6-yr design study, leading to the final selected configuration of a modular wedged design with a partial case, using a beryllium-copper conductor material. D. Ravenscroft (LLNL) proposed the design of a beam-driven standard mirror device to serve as a 14-MeV neutron source for fusion materials testing and development. The neutron wall load of 5 to 10 MW/m^2 would allow accelerated materials testing. L. J. Qiu (IPP Hefei) submitted a description of the conceptual design of the tokamak experimental hybrid reactor. This device would use a JET-like plasma to produce 100 kg/yr of ^{239}Pu in a beryllium-multiplied fission-suppressed blanket. R. R. Peterson (University of Wisconsin) presented the latest version of the University of Wisconsin light ion–driven inertial fusion reactor, LIBRE-LiTE. He proposed use of lithium or lithium-lead as the blanket breeding material, the wetted-wall protection scheme, and the conductor for the ion beam focusing magnets. Significant effort was devoted to the critical details of the light ion beam focusing system using ballistic propagation. D. E. Driemeyer (MDAC) described the development of a parametric systems analysis code for use in ICF reactor design studies. He detailed the application of this code to the conceptual design of two reactors, driven by KrF lasers and heavy ion beams. S. Sharafat (UCLA) presented the reactor design configuration for the ARIES-III D-^{3}He–fueled advanced tokamak reactor.

Modular construction of the first-wall shield modules allows maintenance by vertical lift between the TF coils. The second stability physics regime allows high-beta operation but requires helical kink mode stability control coils around the shield that complicate the maintenance procedure. An innovative segmented design of these coils was proposed.

Plasma Engineering
T. K. Mau (UCLA)

The poster session on plasma engineering covered a wide variety of topics ranging from plasma stability to MHD power conversion to advanced fuels. Significant advances have been made by J. A. Leuer and J. C. Wesley (GA) in analyzing the passive and active vertical stability properties of ITER. Their work was based on a linearized, rigid-body model of plasma motion and covers normal, startup, and off-normal plasma conditions. A passive stabilization system based on the twin-loops concept was proposed, with optimized performance (>100% margin and 20 s^{-1} growth rate under nominal conditions; 45% margin and 56 s^{-1} growth rate in off-nominal conditions). The prescribed one-turn voltage and current limits ($V < 3.0$ kV, $I < 300$ kA) allow the system to control 10- and 5-cm vertical excursions for nominal and off-nominal configurations, respectively.

Continuing his pioneering work on compact fusion advanced Rankine cycle (CFAR) coupled to tokamak reactor plasmas, B. G. Logan (LLNL) considered an entirely new regime (CFAR II) for compact, low-cost MHD conversion, which requires higher pressure (>100 atm) and temperature (>1050 K) metal vapor plasmas. He investigated the generic performance of CFAR II conversion of optically thick, singly ionized plasmas, and the equivalent balance-of-plant (BOP) costs for a variety of working fluids (mixtures of potassium, lithium, aluminum, lead, fluorine, and/or beryllium). High conversion efficiencies (>50%) due to high temperatures (1 to 3 eV) and low BOP costs (1 to 2 mill/kW·h) due to high mass power density [>10 MW(electric)/t)] have been found. This important finding may well lead to a new area of research: that of searching for means to directly couple fusion energy to a suitable Rankine cycle fluid.

A useful algorithm was developed by C. G. Bathke et al. (LANL) for determining equilibrium field (EF) coils from fixed-boundary equilibria in their work related to ARIES-III. The externally produced flux is uniquely specified by requiring minimum current in the EF coils for fixed coil positions. Singular value decomposition is then used to determine the EF coil currents. The coil positions are optimized to minimize simultaneously the stored magnetic energy and the difference between the external flux produced by the EF coils and that required by the equilibrium. This technique will prove useful in future EF coil designs.

Other interesting works in the session that should be mentioned include (a) J. T. Scoville's (GA) design of a correction coil to eliminate the $m = 2$, $n = 1$, field error and to allow for an expanded operating parameter space for DIII-D with proper phasing of its segments; (b) E. Pedretti's (ENEA) calculations of neutron production rates for various D-T startup scenarios to an ignited D-^{3}He plasma; and (c) P. G. Papanikolaou's (Purdue University) development of a one-dimensional code to analyze the global stability of a Vlasov equilibrium.

ICF Driver Technology
L. D. Stewart (GA)

In the area of ICF drivers, the development path to a reactor was traced through papers on the upgrading of an existing induction linac driver (ETA-II), on the design of a target chamber for a laser facility (PHEBUS) anticipating experiments with yields of tens of megajoules, and on the design trades for an induction linac driver for a 1-GW(electric) power plant. Although the immediate application reported by S. M. Hibbs (LLNL) for the upgraded existing linac is as a driver for a 140-GHz tokamak heating system, this electron induction linac continues to generate a design and performance data base with relevance to the heavy-ion beam induction linac approach. Recent changes to enhance performance include a multicable current feed system, kinematic intercell magnet mounts, and improved alignment techniques. M. Rambeau (CELV) stated that a key issue for implementing the near-term target chamber is protection of the final optics. The approach chosen, a sacrificial debris shield with a precharge of low-pressure gas in each beam tube, is of low risk for the planned laser-target experiments but is not reactor relevant. The reactor heavy-ion beam driver design trades reported by L. D. Stewart (GA) indicated that unknowns in final focus physics lead directly to uncertainties in final focus design, but these unknowns do not have a large impact on overall driver design and costs.

The design, fabrication, and operation of an ultraviolet preionized excimer KrF laser amplifier was reported by L. D. Meixler (PPPL). The design output of 5 J has been achieved in a 20-ns pulse at a 248-nm wavelength. This laser is used as a driver for soft X-ray lasers with wavelengths of 15 to 20 nm.

Diagnostic Systems
R. K. Fisher (GA)

T. N. Carlstrom (GA) presented results from the new DIII-D multipulse Thomson scattering diagnostics. The high spatial resolution (1.3 cm) and dynamic range (10 eV to 20 keV) has allowed studies of the steep density and temperature gradients near the plasma edge. It will soon operate in a burst mode with eight separate Nd:YAG lasers to allow high temporal resolution (<10 kHz) as well. The system can be used to study L- to H-mode transitions and ELMs and plasma transport. This is a significant step forward in Thomson scattering.

P. Gohil (GA) described the DIII-D charge-exchange recombination diagnostic system that has high spatial resolution near the plasma separatrix and shows a plasma poloidal rotation and radial electric field profile change at the L- to H-mode direction. Y. Neyatani (JAERI) described the development of magnetic diagnostics for JT-60U that are capable of operating at 500°C.

D. Nilson (LLNL) reported on MTX diagnostic upgrades. A novel technique to rotate a polarizer at up to 120 000 rpm was described. The optical element is mounted inside the shaft of a rotating motor. He also described a collaboration with JAERI to develop a diagnostic to measure the FEL electric field strength using the Stark effect on an injected helium beam.

K. Ritting (UCLA), in describing UCLA's work on far infrared interferometry for the MST at Wisconsin, told of their success in increasing the infrared laser power by switching from a glass laser cavity to a corrugated metal waveguide cavity.

Posters by M. McCarthy (PPPL) and R. S. Granetz (MIT) described the TFTR microwave reflectometer diagnostic and the Alcator C-Mod magnetics instrumentation, respectively. Posters by J. W. Coleman (MIT) and L. D. Meixler

(PPPL) described diagnostics aspects of a neutron source and an X-ray laser, respectively.

Cryogenic Systems
M. L. Tupper (GA)

Cryogenic-cooled vacuum pumps are being designed to be installed into the primary vacuum vessels of both JET and DIII-D. The purpose of these cryopumps is to enhance particle removal from diverted plasmas. Work reported by the GA design team for DIII-D highlighted the cryopump design (J. P. Smith), development and verification of analytical models for both DIII-D and JET (C. B. Baxi), and the design for the external cryosupport system (K. M. Schaubel). G. Celentano (JET) reported on the installation of the JET pumped divertor system inside the vacuum vessel.

Added constraints and requirements for installation into these vacuum chambers has necessitated innovative designs where some standard cryogenic system design approaches were unusable. Heat loading and voltage standoff are two areas of continued concern. The heat loads expected to be incident to these cryopumps is difficult to assess because of uncertainties in the particle energy distribution. Also, the uncertainty as to whether or not ionized particles may be present in these regions leads to concerns over allowable voltage potentials. Interesting data concerning the success of the engineering designs for these in-vessel cryopumps and the physics ramifications of this type of a particle removal technique should be forthcoming.

D. W. Sedgley (Grumman) reported on experiments that demonstrated improved practicality for the cryogenic pumping of helium. The latest experiments have shown that both hydrogen and helium can be pumped onto a single surface of activated charcoal without interfering with the anticipated pumping speed of the helium. This work combined with the development of in-vessel cryopumps may well set the direction for the future development of advanced particle removal systems for fusion reactors.

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