16th ANS Topical Meeting on the Technology of Fusion Energy

Monona Terrace Community & Convention Center
Madison, Wisconsin
September 14-16, 2004

Program & Abstracts

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Forward

Welcome to the 16th Topical Meeting on the Technology of Fusion Energy! We are honored to host this meeting that began in 1972 and was first hosted by General Atomics in San Diego. The meeting has been held approximately every 2 years since then and has chronicled the progress from our early notions of what fusion power plants might look like to the present concepts for next step devices like ITER (the International Thermonuclear Experimental Reactor) and NIF (the National Ignition Facility). The earliest years were dominated by the technologies of magnetically confined fusion plasmas and in more recent times, the emphasis on technologies associated with inertially confined plasmas has increased. Another trend is that the design of conceptual electricity producing power plants, which was so prominent in the 70's and 80's, has gained momentum recently. This work has been augmented with technology development experiments and studies on near term facilities. Alternate applications of fusion energy have received more attention in the past decade and small efforts on the use of fusion energy in space have even begun at various laboratories.

ITER has been the focus of much of the fusion technology research over the past 20 years. There was a short period (1998-2003) when the United States effort was reduced but it now appears that efforts are increasing in that area as the international community moves towards a site selection decision. The construction of NIF is nearly complete as we hold this conference and the NIF facility contains many components produced by the community that has used TOFE as its vehicle of information exchange. The Technical Program Committee has tried to mix both MFE and IFE papers so that both communities can learn what is happening in the other's research area.

We hope that you enjoy your stay in Madison and look forward to exchanging technical information with our colleagues from the United States and abroad. The goal of producing clean, safe, and economic energy from fusion is a noble task and we trust that it will be eventually achieved by many of the attendees of this conference.

Gerald L. Kulcinski
University of Wisconsin
General Chair

Acknowledgement and Appreciation

The 16th Topical Meeting on the Technology of Fusion Energy (TOFE) is hosted by the Fusion Technology Institute at the University of Wisconsin and co-sponsored by the American Nuclear Society, the Atomic Energy Society of Japan, the U.S. Department of Energy, the University of Wisconsin, and the Wisconsin Chapter of the American Nuclear Society.

The organizers express their heartfelt appreciation to those who have contributed financially to this meeting. The list includes, at press time, the Naval Research Laboratory, General Atomics, Sandia National Laboratory, Oak Ridge National Laboratory, the Department of Energy, the Atomic Energy Society of Japan, AREVA, and the Fusion Technology Institute. We also thank the ANS Fusion Energy Division for providing funds to defray some of the travel costs for several students to attend the TOFE meeting.

Organizing Committee

General Chair	Gerald Kulcinski (UW)
Vice Chair	Masahiro Seki (JAERI)
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Special Events and Guest Program	John Santarius (UW)
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Chair:	Laila El-Guebaly (UW)
Assistant Chairs:	Ichiro Yamamoto (Nagoya U.)
	René Raffray (UCSD)

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M. Enoeda	JAERI-J
G. Federici	ITER-EU
Y. Gohar	ANL
D. Goodin	GA
A. Hassanein	ANL
D. Johnson	PPPL
A. Kohyama	Kyoto U.-J
TK. Mau	UCSD
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W. Meier	LLNL
K. Mima	Osaka U.-J
N. Morley	UCLA
M. Nishikawa	Kyushu U.-J
C. Olson	SNL
P. Peterson	UCB
R. Peterson	LANL
A. Sagara	NIFS-J
J. Santarius	UW
M. Sawan	UW
K. Schultz	GA
J. Sethian	NRL
L. Snead	ORNL
M. Sugimoto	JAERI-J
I. Sviatoslavsky	UW
N. Taylor	UKAEA-EU
P. Titus	MIT
L. Waganer	Boeing
P. Wilson	UW
S. Zinkle	ORNL

Schedule

	Monday 9 / 13 / 04	Tuesday 9 / 14 / 04	Wednesday 9 / 15 / 04	Thursday 9 / 16 / 04
7 - 8 AM		Continental Breakfast Buffet	Continental Breakfast Buffet	Continental Breakfast Buffet
8 - 10 AM		Plenary (Opening Statement + 4 talks)	Plenary (4 talks)	Oral (3 // sessions, 6 talks per session)
10 - 10:30		Coffee Break	Coffee Break	Coffee Break
10:30 - 12		Oral (3 // sessions, 5 talks per session)	Oral (3 // sessions, 5 talks per session)	Plenary (3 talks + wrap-up talk)
12 - 1:30		Lunch	Lunch	Lunch
1:30 - 3:30 PM		Posters + Exhibits Refreshments	Posters + Exhibits Refreshments	Technical Tours 1:30 - 4:30 PM ARIES Project Meeting 1 - 5 PM - Hall of Ideas-H
3:30 - 5:30		Oral (3 // sessions, 6 talks per session)	Oral (3 // sessions, 6 talks per session)	
Evening	Registration & Mixer 6 - 9	Reception 5:30 - 7:30	Cash Bar: 6 - 7 Banquet / Awards 7 - 10	

Program Summary and TOFE Activities

Tuesday September 14, 2004

Tuesday Morning – 7-8 AM – Continental Breakfast Buffet **– Grand Terrace**
Tuesday Morning – 8-10 AM
Plenary-I – Fusion Program Overviews **– Lecture Hall**
Coffee Break – 10-10:30 AM – Grand Terrace
Tuesday Morning – 10:30-12 AM
Oral-I-1 – Engineering of Experimental Devices **– Lecture Hall**
Oral-I-2 – High Average Power Laser - Special Session **– Hall of Ideas-H**
Oral-I-3 – Socioeconomics, Safety, Radwaste, and Licensing **– Hall of Ideas-E**
Lunch – 12–1:30 PM – See "Map-Central" for list of nearby restaurants
Poster Session–I – 1:30-3:30 PM – Grand Terrace
Code Development, Testing, and Validation Experiments
Target Development and IFE Technology
Nuclear Analysis and Experiments
In-Vessel Components and Power Conversion
Materials Development
Magnets and Structural Analysis
Refreshments – 1:30-3:30 – Grand Terrace
Tuesday Afternoon – 3:30 – 5:30 PM
Oral-I-4 – Power Plant Studies **– Lecture Hall**
Oral-I-5 – ITER Test Blanket Modules - Special Session **– Hall of Ideas-H**
Oral-I-6 – Non-Electric Applications **– Hall of Ideas-E**
Reception – 5:30-7:30 PM – Grand Terrace

Wednesday September 15, 2004

Wednesday Morning – 7-8 AM – Continental Breakfast Buffet **– Grand Terrace**
Wednesday Morning – 8-10 AM
Plenary-II – Fusion Development and Near-Term Facilities **– Lecture Hall**
Coffee Break – 10-10:30 AM – Grand Terrace
Wednesday Morning – 10:30-12 AM
Oral-II-1 – ARIES Compact Stellarator - Special Session **– Lecture Hall**
Oral-II-2 – Target Development and IFE Technology **– Hall of Ideas-H**
Oral-II-3 – Latest Technology and Tritium System **– Hall of Ideas-E**
Lunch – 12–1:30 PM -- See "Map-Central" for list of nearby restaurants
Poster Session–II – 1:30-3:30 PM – Grand Terrace
Plasma Control and Diagnostics
Non-Electric Applications and IEC Research
Safety and Environment
High Flux Components and Chamber Clearing
Thermal and MHD Analyses
Refreshments – 1:30-3:30 – Grand Terrace

Wednesday Afternoon – 3:30 – 5:30 PM

Oral-II-4 – US Contributions to ITER - Special Session – Lecture Hall

Oral-II-5 – Breeding Blanket Development – Hall of Ideas-H

Oral-II-6 – IFE Designs and Technology – Hall of Ideas-E

Cash Bar – 6-7 PM – Community Terrace

Banquet and Awards – 7-10 PM – Community Terrace

Thursday September 16, 2004

Thursday Morning – 7-8 AM – Continental Breakfast Buffet – Grand Terrace

Thursday Morning – 8-10 AM

Oral-III-1 – Materials Development – Lecture Hall

Oral-III-2 – High Heat Flux Components – Hall of Ideas-H

Oral-III-3 – Nuclear Technology Experiments and Testing – Hall of Ideas-E

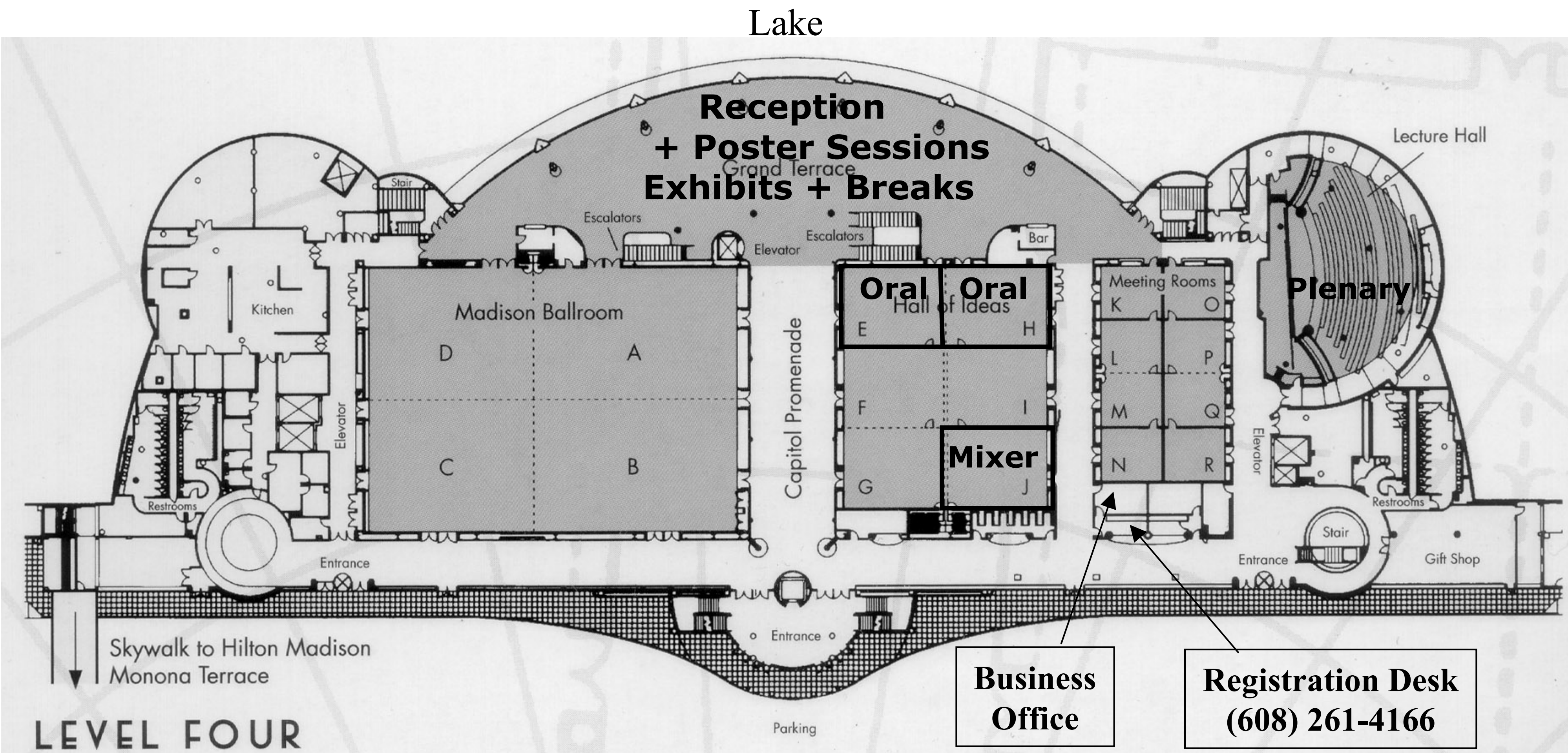
Coffee Break – 10-10:30 AM – Grand Terrace

Thursday Morning – 10:30-12 AM

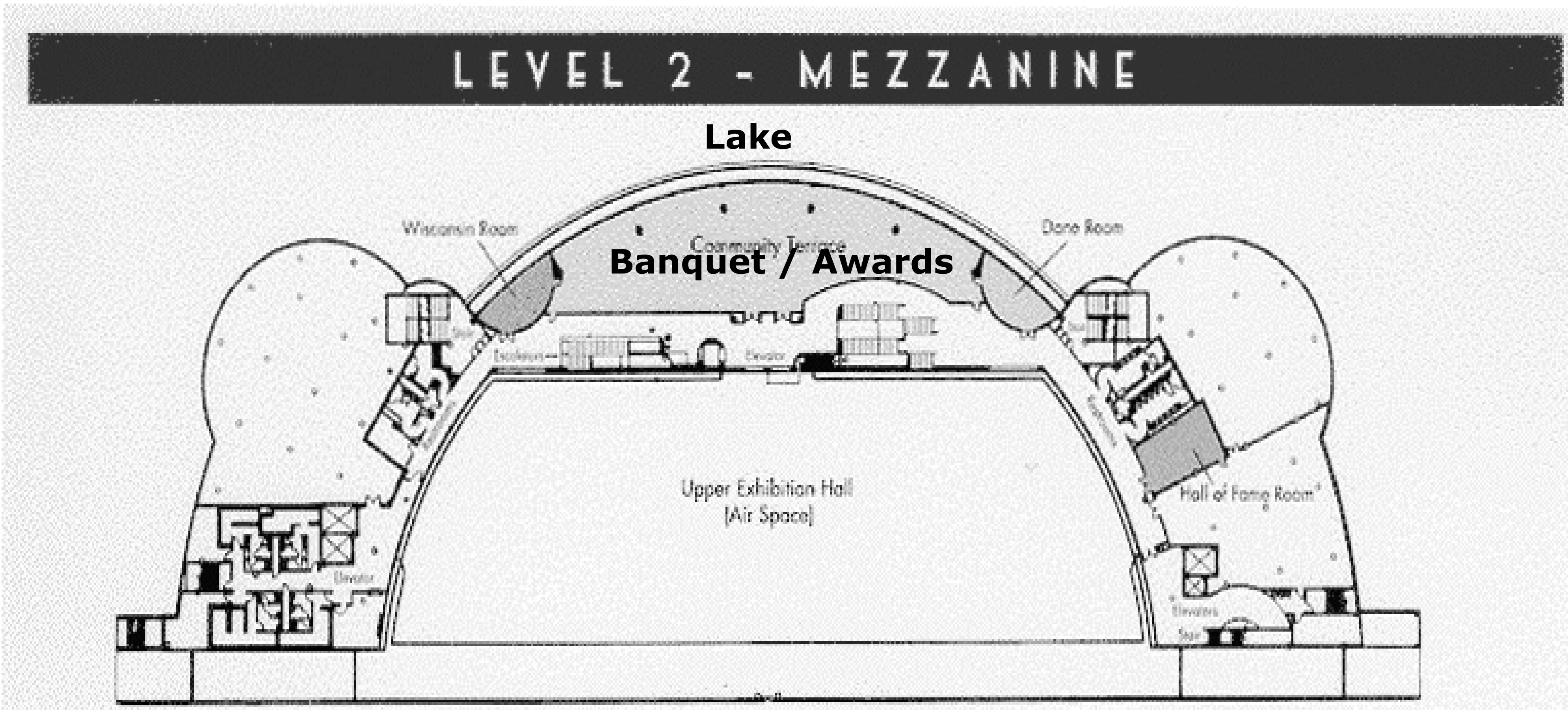
Plenary-III – The ITER Project – Lecture Hall

Thursday Afternoon – 1:30-4:30 PM – Technical Tours

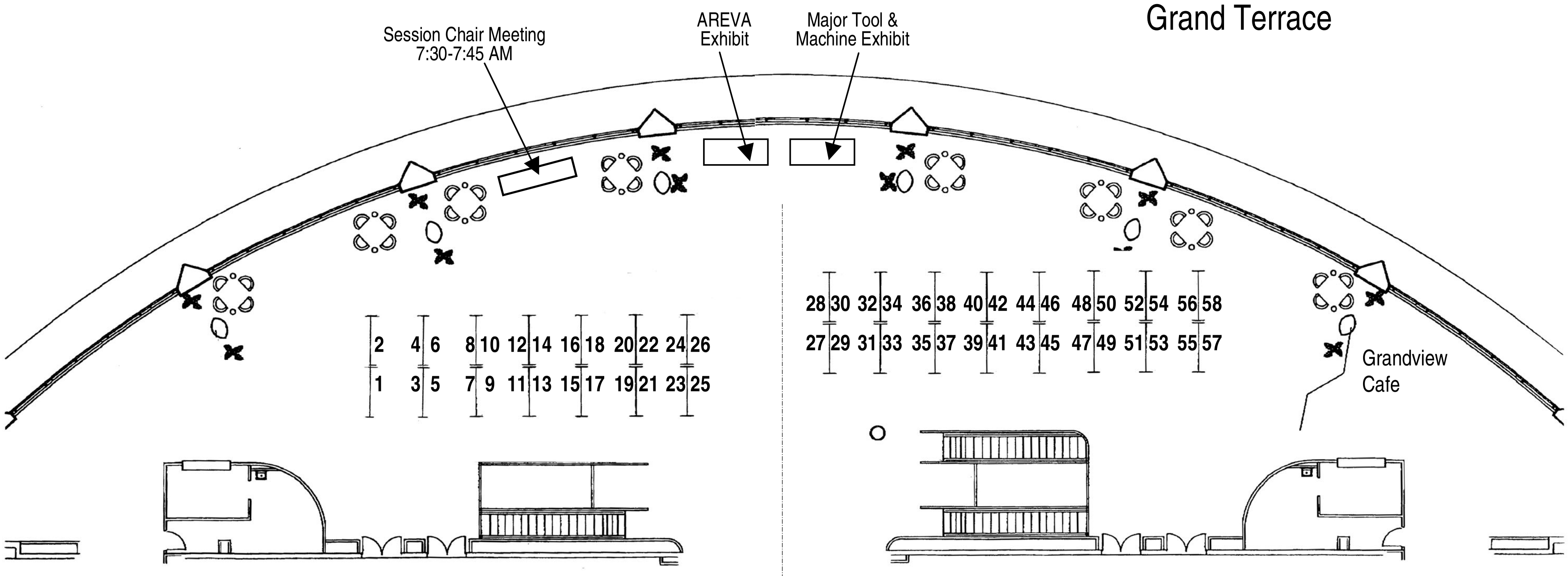
Meeting Rooms
(4th Floor)



Banquet / Awards
(2^{nd} Floor)



Posters/Exhibits Layout



16^{th} TOFE Technical Program

Tuesday, September 14, 2004

Tuesday Morning - 8-10 AM (5 talks, 15, 30, or 25 min each)

Fusion Program Overviews (Kulcinski) - Lecture Hall

	8:00- 8:15	Kulcinski	Opening Statement and Program Overview
PL-I-1	8:15-8:45	Schmitt	Large Energy Development Projects: Lessons Learned from Space and Politics
PL-I-2	8:45-9:10	Davies	US MFE National Picture
PL-I-3	9:10-9:35	Keane	Status of the US Inertial Confinement Fusion Program
PL-I-4	9:35-10:00	Seki	Overview of Recent Japanese Activities in Fusion Technology

Coffee Break - 10-10:30 AM - Grand Terrace

Tuesday Morning - 10:30-12 AM (3 || sessions, 5 talks per session, 18 min each)

Engineering of Experimental Devices (Meade, Kellman) - Lecture Hall

O-I-1.1	10:30-10:48	Kellman	Advanced Control Techniques and High Performance Discharges on DIII-D
O-I-1.2	10:48-11:06	Gasparotto	W7-X Progress
O-I-1.3	11:06-11:24	Fonck	Recent progress of Low Aspect Ratio Machines
O-I-1.4	11:24-11:42	Kaye	Progress in Technology at JET
O-I-1.5	11:42-12:00	Simmons	Recent Progress in NCSX Design and Fabrication

High Average Power Laser - Special Session (Sethian, Blanchard) - Hall of Ideas-H

O-I-2.1	10:30-10:48	Sethian	The Path to Develop Laser Fusion Energy
O-I-2.2	10:48-11:06	Jones	KrF Laser Drivers for Initial Fusion Energy
O-I-2.3	11:06-11:24	Bibeau	Diode-Pumped Solid-State Laser Driver for Inertial Fusion Energy
O-I-2.4	11:24-11:42	Dragojlovic	Effects of Chamber Geometry and Gas Properties on Hydrodynamic Evolution of IFE Chambers
O-I-2.5	11:42-12:00	Blanchard	Development of a Dry Wall Concept for Laser IFE Chamber

Socioeconomics, Safety, Radwaste, and Licensing (Taylor, Sheffield) - Hall of Ideas-E

O-I-3.1	10:30-10:48	Sheffield	Future World Energy Demand and Supply: China and India and the Potential Role of Fusion Energy
O-I-3.2	10:48-11:06	Petti	Status of Safety and Environmental Activities in the US Fusion Program
O-I-3.3	11:06-11:24	El-Guebaly	Evolution of Clearance Standards and Implications for Radwaste Management of Fusion Power Plants
O-I-3.4	11:24-11:4	Taylor	Key Issues for the Safety and Licensing of Fusion
O-I-3.5	11:42-12:00	D'haeseleer	Fusion Power; a Strategic Choice for the Future Energy Provision. Why is so Much Time Wasted for Decision Making?

Lunch - 12-1:30 PM

Poster Session-I - 1:30 - 3:30 PM - Grand Terrace

Code Development, Testing, and Validation Experiments (Morley, Sugimoto)

P-I-1	Shestakov	In-pile Assemblies for Testing of Li2TiO3 Lithium Ceramic Blanket
P-I-2	Tazhibayeva	KTM Experimental Complex Project Status
P-I-3	Nygren	Thermal Modeling of the Sandia Flinabe (LiF-BeF2-NaF) Melt Experiment
P-I-4	Maebara	Power-balance control by Slug Tuner for the 175MHz RFQ linac in IFMIF project
P-I-5	McDonald	Measurement of the Melting Point Temperature of Several Lithium-Sodium-Beryllium Fluoride Salt (FLiNaBe) Mixtures
P-I-6	Ranjan	Investigation of hydrodynamic instabilities in shock-accelerated flows for ICF
P-I-7	Debonnel	Visual Tsunami: A versatile, user-friendly radiation hydrodynamics design code
P-I-8	Debonnel	Validation of the Tsunami code through the Condensation Debris Experiment
P-I-9	Debonnel	Practical ablation models for IFE chamber design

Target Development and IFE Technology (Petzoldt, Wurden)

P-I-10	Petzoldt	Target Injection Tracking and Position Prediction Update
P-I-11	Vermillion	Fabrication of a Full Density Polyvinylphenol Overcoat for IFE Targets
P-I-12	Frey	Rep-Rated Target Injection for Inertial Fusion Energy
P-I-13	Olson	Target Physics Scaling for Z-Pinch Inertial Fusion Energy
P-I-14	Rose	Large-Area Electron Beam Diode and Gas Cell Design for a KrF Laser IFE System
P-I-15	Wu	Electron Injection for Space - Charge Contron in HIBF
P-I-16	Calderoni	Experimental study of voltage breakdown over flibe liquid surfaces for IFE applications
P-I-17	Hahn	Characterization of Arc Generated Plasma Interactions with a Liquid Metal Medium
P-I-18	Bardet	Liquid Vortex Shielding for Fusion Energy Applications
P-I-19	Christensen	Thermal Loading of a Direct-Drive Target in Rarefied Gas
P-I-20	Christensen	Modeling DT Vaporization and Melting in a Direct-Drive Target

Nuclear Analysis and Experiments (Youssef, Sato)

P-I-21	Muroga	Neutronics Investigation into Lithium/Vanadium Test Blanket Modules
P-I-22	Yamauchi	Estimation of Radioactivities in the IFMIF Liquid Lithium Loop due to the Erosion and Corrosion of Target Back-wall
P-I-23	Sato	Neutronics experiments using small partial mockup of the ITER test blanket module with solid breeder
P-I-24	Tanaka	Tritium Self-Sufficiency and Neutron Shielding Performance of Liquid Li Self-Cooled Helical Reactor
P-I-25	El-Guebaly	Views on Neutronics and Activation Issues Facing Thick Liquid-Protected IFE Chambers
P-I-26	Youssef	Activation Analysis for Two Molten Salt Dual-Coolant Blanket Concepts for the US Demo Reactor
P-I-27	Youssef	On the Strategy and Requirements for Neutronics Testing in ITER
P-I-28	Wang	Three-Dimensional Modeling of Complex Fusion Devices Using CAD-MCNP Interface

In-Vessel Components and Power Conversion (Gohar, Enoeda)

P-I-29	Gohar	Blanket Design and Optimization Demonstrations of the First Wall/Blanket/Shield Design and Optimization System
P-I-30	Sviatoslavsky	A Self-Cooled Lithium Blanket Concept for the HAPL Conceptual Laser IFE Power Plant
P-I-31	Wang	Maintenance Approaches for ARIES-CS Power Core
P-I-32	Rochau	Manufacturing Concepts for an IFE Power Plant Using Z-Pinch Technology
P-I-33	Viola	National Compact Stellarator Experiment (NCSX) Vacuum Vessel Manufacture
P-I-34	Takeno	Experiments to Improve Power Conversion Paramters in a TWDEC Simulator
P-I-35	Yasaka	Experimental Simulation on Particle Discrimination for Direct Energy Conversion
P-I-36	Zhao	Optimized Helium-Brayton Power Conversion for Fusion Energy Systems

Materials Development (Diegele, Sharafat)

P-I-37	Tsuchiya	Radiation Induced Conductivity of Proton Conductive Ceramics
P-I-38	Wakai	Effect of Initial Heat Treatment on DBTT of F82H Steel Irradiated by Neutrons
P-I-39	Nagasaka	Tungsten Coating on Low Activation Vanadium Alloy by Plasma Spray Process
P-I-40	Maday	Hydrogen Embrittlement Susceptibility of Conventional and Reduced Activation 9Cr Steels
P-I-41	Ogiwara	Synergistic Influence of Displacement Damage and Helium/dpa on Microstructural Evolution and Radiation-Induced Hardening of Reduced Activation Ferritic/Martensitic Steel
P-I-42	Sawada	Corrosion Behavior of Insulator Coatings for Fusion Reactor Lithium Blankets
P-I-43	Ozawa	Cavity Swelling Behavior in SiC/SiC under Charged Particle Irradiation
P-I-44	Sharafat	An Innovative Solid Breeder Material for Fusion Applications
P-I-45	Hashimoto	Helium Retention of Ion-irradiated and annealed Tungsten Foils
P-I-46	Sharafat	Elasto-plastic Dislocation-based Constitutive Modeling within Full Scale 3-D Thermo-mechanical FEM Analysis of an ITER-TBM

Magnets and Structural Analysis (Titus, Simmons)

P-I-47	Senkowicz	Upper Critical Field Improvement in MgB2 by Mechanical Alloying
P-I-48	Hashizume	Jointing Performance in HTc SC Tape for remountable magnet system
P-I-49	Bromberg	Advanced Options for Modular Stellarator Magnets
P-I-50	Jewell	Sn content and alloying effects in ITER Nb3Sn strand
P-I-51	Titus	Seismic Analysis of the National Compact Stellarator (NCSX)

P-I-52	Freudenberg	Non-linear Analysis of the Modular Coil Windings for the National Compact Stellarator Experiment
P-I-53	Myatt	Electromagnetic Linear Structural Analysis of the National Compact Stellarator Experiment (NCSX) Modular Coil System
P-I-54	Kozub	Testing of NCSX Composite Coil Material Properties
P-I-55	Titus	Structural Analysis of the JET TAE Antenna
P-I-56	Williamson	Wire Debris Modeling of the Z-Accelerator
P-I-57	Dahlgren	Structural Analysis of the NCSX Vacuum Vessel
P-I-58	Doshi	Design Requirement, Qualification Tests and Integration of a Thin Solid Lubricant Film of MoS2 for Cold Mass Support Structure of the Steady State Superconducting Tokamak SST-1

Tuesday Afternoon - 3:30 - 5:30 PM (3 || sessions, 6 talks per session, 20 min each)

Power Plant Studies (Waganer, Rochau) - Lecture Hall

O-I-4.1	3:30-3:50	Najmabadi	ARIES-IFE Assessment of Operational Windows for IFE Power Plants
O-I-4.2	3:50-4:10	Cook	European Fusion Power Plant Studies
O-I-4.3	4:10-4:30	Wilson	Synergies between the Generation IV and Advanced Fusion Power Plants
O-I-4.4	4:30-4:50	Meade	FIRE, A Test Bed for ARIES-RS/AT Advanced Physics and Plasma Technology
O-I-4.5	4:50-5:10	Rochau	Progress Toward Development of an IFE Power Plant Using Z-Pinch Technology
O-I-4.6	5:10-5:30	Dolan	Helical Fusion Power Plant Economics Studies

ITER Test Blanket Modules - Special Session (Abdou, Sawan) - Hall of Ideas-H

O-I-5.1	3:30-3:50	Chuyanov	Interface of Blanket Testing and ITER design
O-I-5.2	3:50-4:10	Abdou	Blanket Testing Issues and Requirements
O-I-5.3	4:10-4:30	Boccaccini	European Test strategy for Test blanket modules to be tested in ITER
O-I-5.4	4:30-4:50	Enoeda	Plans and Strategy for ITER Blanket Testing in Japan
O-I-5.5	4:50-5:10	Morley	Thermofluid Magnetohydrodynamic Issues for Liquid Breeders
O-I-5.6	5:10-5:30	Ying	Engineering Scaling Requirements for Solid Breeder Blanket Testing

Non-Electric Applications (Schultz, Santarius) - Hall of Ideas-E

O-I-6.1	3:30-3:50	Schultz	Fusion Production of Hydrogen; How Fusion Energy Can Fuel the Hydrogen Economy
O-I-6.2	3:50-4:10	Konishi	Potential Fusion Market for Hydrogen Production Under Environmental Constraints
O-I-6.3	4:10-4:30	Stacey	Tokamak Neutron Source Based Spent Nuclear Fuel Transmutation Reactors
O-I-6.4	4:30-4:50	Yoshikawa	Research and Development of Landmine Detection System by a Compact Fusion Neutron Source
O-I-6.5	4:50-5:10	Miley	RF Ion Source-Driven IEC Design and Operation
O-I-6.6	5:10-5:30	Santarius	Overview of University of Wisconsin Inertial-Electrostatic Confinement Fusion Research

Wednesday, September 15, 2004

Wednesday Morning - 8-10 AM (4 talks, 30 min each)

Fusion Development and Near-Term Facilities (El-Guebaly) - Lecture Hall

PL-II-1	8:00-8:30	Dean	Historical Perspective on the United States Fusion Program
PL-II-2	8:30-9:00	Iotti	The Role of Industry in Fusion Development
PL-II-3	9:00-9:30	Wuest	The National Ignition Facility: Laser Performance and First Experiments
PL-II-4	9:30-10:00	Andreani	European Technological Effort in Preparation for ITER Construction

Coffee Break - 10-10:30 AM - Grand Terrace

Wednesday Morning - 10:30-12 AM (3 || sessions, 5 talks per session, 18 min each)

ARIES Compact Stellarator - Special Session (Najmabadi, Cook) - Lecture Hall

O-II-1.1	10:30-10:48	Najmabadi	Exploration of Compact Stellarators as Power Plants: Initial Results from ARIES-CS Study
O-II-1.2	10:48-11:06	Garabedian	Reactors with Stellarator Stability and Tokamak Transport
O-II-1.3	11:06-11:24	Lyon	Optimization of Stellarator Reactor Parameters
O-II-1.4	11:24-11:42	Raffray	Attractive Design Approaches for a Compact Stellarator Power Plant
O-II-1.5	11:42-12:00	El-Guebaly	Benefits of Radial Build Minimization and Requirements Imposed on ARIES-CS Stellarator Design

Target Development and IFE Technology (Goodin, Peterson) - Hall of Ideas-H

O-II-2.1	10:30-10:48	Goodin	Demonstrating a Target Supply for Inertial Fusion Energy
O-II-2.2	10:48-11:06	Norimatsu	Development of target fabrication and injection for Laser Fusion in Japan
O-II-2.3	11:06-11:24	Moses	High Energy Density Simulations for IFE Reactor Design
O-II-2.4	11:24-11:42	Callahan	Target Designs for Heavy Ion Inertial Fusion Energy
O-II-2.5	11:42-12:00	Streit	Fabrication of Overcoated Divinylbenzene (DVB) Shells

Latest Technology and Tritium System (Dean, Cadwallader) - Hall of Ideas-E

O-II-3.1	10:30-10:48	Hosogane	Recent Technological Progress for Advanced Tokamak Research in JT-60U and JFT-2M
O-II-3.2	10:48-11:06	Lucon	An Integrated Approach to Fusion Material Research at SCK-CEN
O-II-3.3	11:06-11:24	Cadwallader	Comparison of Tritium Component Failure Rate Data
O-II-3.4	11:24-11:42	Burruss	Development in Remote Collaboration and Distributed Computing
O-II-3.5	11:42-12:00	Putvinskaya	Information Technology Systems for Fusion Industry and ITER project

Lunch - 12-1:30 PM

Poster Session-II - 1:30 - 3:30 PM - Grand Terrace

Plasma Control and Diagnostics (Mau, Hosogane)

P-II-1	Machkour	Magnetic method To characterize the Current Densities in a Breaker Arc
P-II-2	Scoville	Automatic Fault-Checking System on the DIII-D Tokamak
P-II-3	Oliva	Four Barrel Pellet Injector Upgrade on the Madison Symmetric Torus (MST)
P-II-4	Kowbel	Hybrid AL/SiC composite optics for IFE applications
P-II-5	Cengher	EBW coupling using a twin waveguide launcher on the MST reversed field pinch
P-II-6	Matsuura	Effect of Nuclear Elastic Scattering on Neutral Beam Injection Heating in Thermonuclear Plasmas
P-II-7	Deranian	Integrated Plasma Control in Next-Generation Devices Using DIII-D Modeling and Simulation Approaches
P-II-8	Akino	Long pulse operation of NBI system for JT-60U
P-II-9	Walker	Nonlinear Methods for Current Limit Constraint Satisfaction in Tokamak Plasma Shape Control
P-II-10	Welander	Nonrigid, Linear Plasma Response Model Based on Perturbed Equilibria for Axisymmetric Tokamak Control Design
P-II-11	Ebara	Numerical Study on behavior of hydrogen ice pellet in drift tube
P-II-12	Mau	Alpha Particle Loss and Heat Load Assessment for Compact Stellarator Reactors
P-II-13	Kamberov	Correlations of ELM frequency with pedestal plasma parameters

Non-Electric Applications and IEC Research (Nebel, Konishi)

P-II-14	Cheng	Performance Characteristics of Actinide-Burning Fusion Power Plants
P-II-15	Cipiti	The Production of 13N using Beam-Target D-3He Fusion
P-II-16	Takamatsu	Improvement of an Inertial Electrostatic Confinement Fusion Device
P-II-17	Osawa	Optimal Position of Ion Source for High Performance of IEC
P-II-18	Osawa	Numerical Study on Hollow Cathode Discharge of IEC Fusion
P-II-19	Ueno	Measurement of Ion Energy Distributions in a Cylindrical Inertial Electrostatic Confinement Fusion (C-IECF) Device
P-II-20	Yoshinaga	Fast Neutral Generation by Charge Exchange Reaction and its Effect on Nuclear Burning in Inertial Electrostatic Confinement Fusion Systems
P-II-21	Noborio	One Dimensional Simulation of an Inertial Electrostatic Confinement Fusion at Low Gas Pressure Operation
P-II-22	Yamamoto	Influence of the Electrode Spacing on the Performance Characteristics of Inertial Electrostatic Confinement Fusion in Low Pressure Operation
P-II-23	Yamauchi	Pulsed Operation of a Compact Fusion Neutron Source Using a High-Voltage Pulse Generator Developed for Landmine Detection
P-II-24	Radel	Implantation of D+ and He+ in Candidate Fusion First Wall Materials
P-II-25	Radel	Optimizing Neutron Production Rates from D-D Fusion in an Inertial Electrostatic Confinement Device
P-II-26	Murali	D-3He Proton Energy Distribution from an IEC Device
P-II-27	Piefer	Design of an Ion Source for Low Pressure IEC Operation
P-II-28	Nebel	An Electrostatic Confinement Experiment to Explore the Periodically Oscillating Plasma Sphere

Safety and Environment (Cheng)

P-II-29	El-Guebaly	Initial Activation Assessment for ARIES Compact Stellarator Power Plant
P-II-30	Cadwallader	Compressed Gas Safety for Experimental Fusion Facilities
P-II-31	Savercool	Activities of the US-Japan Safety Monitor Joint Working Group
P-II-32	Cadwallader	The Worker Exposure Failure Modes and Effects Analysis

High Flux Components and Chamber Clearing (Linke, Mima)

P-II-33	Shin	Design Constraints for Liquid-Protected Divertors
P-II-34	Hirooka	Particle Control by Lithium-Gettered Moving-Surface PFCs in Steady State Magnetic Fusion Devices
P-II-35	Karditsas	Optimization of the HETS He-cooled divertor concept:Thermal-fluid and structural analysis
P-II-36	Morikawa	Experimental Analysis of Soaker Hose Concept for First Wall/Diverter Application
P-II-37	Latkowski	Pulsed X-Ray Exposures and Modeling for Tungsten as an IFE First Wall Material
P-II-38	Whyte	DIONISOS: A new experiment on the dynamics of plasma-surface interactions
P-II-39	Stubbers	The Flowing Liquid-Metal Retention Experiment (FLIRE) Results
P-II-40	Durbin	Impact of Boundary-Layer Cutting on Free-Surface Behavior in Turbulent Liquid Sheets
P-II-41	Durbin	Flow Conditioning Design in Thick Liquid Protection
P-II-42	Debonnel	Visual Tsunami modeling of ballistic and assisted-pinch heavy-ion target chambers

Thermal and MHD Analyses (Karditsas, Sagara)

P-II-43	Novak	Experimental and Numerical Investigation of Mist Cooling for the Electra Hibachi
P-II-44	Yin	Prediction of Pressure Drop in the ITER Divertor Cooling Channels
P-II-45	Pampin-Garcia	Heat transfer issues in finite element analysis of PPCS model bounding accidents
P-II-46	Chiba	Experimental Research on Heat Transfer Enhancement for High Prandtl-Number Fluid
P-II-47	Mignot	Initial Study of Supercritical Fluid Blowdown
P-II-48	Okumura	Evaluation of flow structure in packed-bed tube by visualization experiment
P-II-49	Togashi	Heat Transfer Enhancement Technique with Copper Fiber Porous Media
P-II-50	Modesto-Beato	Thermal Analysis of the Z-Pinch Power Plant Concept
P-II-51	Luo	Modeling Development for Free Surface Flow with Phase Change
P-II-52	An	Experimental & Numerical Study of Ceramic Breeder Pebble Bed Thermal Deformation Behavior
P-II-53	Abou-Sena	Effects of Pulsed Operation Conditions on Effective Thermo-physical Properties of Ceramic Breeder Pebble Beds
P-II-54	Dritselis	Magnetohydrodynamic Turbulent Channel Flow with Transverse Square Cylinders
P-II-55	Smolentsev	MHD Effects on Heat Transfer in a Molten Salt Blanket
P-II-56	Narula	Study of Liquid Metal Film Flow Characteristics under Fusion Relevant Magnetic Fields

Wednesday Afternoon - 3:30 - 5:30 PM (3 || sessions, 6 talks per session, 20 min each)

US Contributions to ITER - Special Session (Baker, Ulrickson) - Lecture Hall

O-II-4.1	3:30-3:50	Antaya	The ITER CS Magnet System
O-II-4.2	3:50-4:10	Ulrickson	Proposed US Participation in Fabrication of the First Wall and Shield for ITER
O-II-4.3	4:10-4:30	Rasmussen	ITER Ion Cyclotron Heating and Fueling Systems
O-II-4.4	4:30-4:50	Vernon	ITER ECH System and US ECH Program for ITER
O-II-4.5	4:50-5:10	Petti	Future US ITER Safety Studies
O-II-4.6	5:10-5:30	Paffett	ITER Plasma Exhaust Processing System

Breeding Blanket Development (Raffray, Sviatoslavsky) - Hall of Ideas-H

O-II-5.1	3:30-3:50	Enoeda	Development of Solid Breeder Blanket at JAERI
O-II-5.2	3:50-4:10	Fischer	EU Blanket Design Activities and Neutronics Support Efforts
O-II-5.3	4:10-4:30	Wong	Assessment of Liquid Breeder First Wall and blanket Options for the DEMODesign
O-II-5.4	4:30-4:50	Sawan	Neutronics Assessment of Molten Salt Breeding Blanket Design Options
O-II-5.5	4:50-5:10	Sagara	Innovative Liquid Blanket Design Activities in Japan
O-II-5.6	5:10-5:30	Raffray	Ceramic Breeder Blanket for ARIES-CS

IFE Designs and Technology (Meier, Olson) - Hall of Ideas-E

O-II-6.1	3:30-3:50	Olson	Development Path for Z Pinch IFE

O-II-6.2	3:50-4:10	Meier	Update on Progress and Challenges in the Development of Heavy Ion Fusion
O-II-6.3	4:10-4:30	Wurden	Overview of Magneto-Inertial Fusion
O-II-6.4	4:30-4:50	Mima	Present Status of Fast Ignition Research and Prospects of FIREX Project
O-II-6.5	4:50-5:10	Yu	The Modular Approach to Heavy-Ion Fusion
O-II-6.6	5:10-5:30	Peterson	Dynamics of Liquid-Protected Fusion Chambers

Thursday, September 16, 2004

Thursday Morning - 8-10 AM (3 || sessions, 6 talks per session, 20 min each)

Materials Development (Zinkle, Kohyama) - Lecture Hall

O-III-1.1	8:00-8:20	Zinkle	Overview of the US Fusion Materials Sciences Program
O-III-1.2	8:20-8:40	Diegele	European Fusion Materials Research Program - Recent Results and Future Strategy
O-III-1.3	8:40-9:00	Kohyama	Recent Accomplishments and Future Prospects of Materials R & D in Japan
O-III-1.4	9:00-9:20	Larbalestier	New Superconductors for Fusion Magnets
O-III-1.5	9:20-9:40	Konys	Status of Tritium Permeation Barrier Development in the EU
O-III-1.6	9:40-10:00	Pint	Recent Progress Addressing Compatibility Issues Relevant to Fusion Environments

High Heat Flux Components (Seki, Hassanein) - Hall of Ideas-H

O-III-2.1	8:00-8:20	Brooks	Overview of ALPS Program
O-III-2.2	8:20-8:40	Linke	EU Developments of High Heat-flux Components
O-III-2.3	8:40-9:00	Hassanein	Plasma/Liquid-Metal Interactions during Tokamak Operation
O-III-2.4	9:00-9:20	Snead	IFE First Wall Survival: Development and Testing of a Refractory Armored Ferritic
O-III-2.5	9:20-9:40	Mirnov	The Use of Ga and Li as Limiter Materials in T-3M and T-11M Tokamaks
O-III-2.6	9:40-10:00	Renk	Laser Inertial Fusion Dry-Wall Materials Exposure to X-rays and Ions

Nuclear Technology Experiments and Testing (Abdel-Khalik, Anderson) - Hall of Ideas-E

O-III-3.1	8:00-8:20	Peng	Engineering and Physics Assessments of Spherical Torus Component Test Facility
O-III-3.2	8:20-8:40	Abdel-Khalik	An Overview of the Fluid Dynamics Aspects of Liquid Protection Schemes for Fusion Systems
O-III-3.3	8:40-9:00	Sugimoto	Recent Progress of Design & Development of IFMIF Activities
O-III-3.4	9:00-9:20	Morley	Progress on Liquid Metal MHD Free Surface Flow Modeling and Experiments
O-III-3.5	9:20-9:40	Anderson	Protection of IFE First Wall Surfaces from Impulsive Loading by Multiple Liquid Layers
O-III-3.6	9:40-10:00	Rodriguez	Z-Pinch Power Plant Shock Mitigation Experiments and Analysis

Coffee Break - 10-10:30 AM - Grand Terrace

Thursday Morning - 10:30-12 AM (3 talks, 30 min each)

The ITER Project (Callen) - Lecture Hall

PL-III-1	10:30-11:00	Barabaschi	ITER Status
PL-III-2	11:00-11:30	Sauthoff	US ITER Project Activities
PL-III-3	11:30-12:00	Baker	Relation of US VLT Program to ITER
-	12:00-12:10	Kulcinski	Wrap-up

Tuesday

September 14, 2004

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Plenary Session I

Fusion Program Overviews

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Large Energy Development Projects: Lessions Learned from Space and Politics

Harrison H. Schmitt

Apollo 17 Astronaut, former U.S. Senator

Harrison H. Schmitt, a native of Silver City, NM, has the diverse experience of a geologist, pilot, astronaut, administrator, businessman, writer, and U.S. Senator. He received his B.S. from Caltech, studied as a Fulbright Scholar at Oslo, and attended graduate school at Harvard. His Ph.D. in geology in 1964 is based on geological field studies in Norway. As a civilian, Schmitt received Air Force jet pilot wings in 1965 and Navy helicopter wings in 1967.

Selected for the Scientist-Astronaut program in 1965, Schmitt organized the lunar science training for the Apollo Astronauts, represented the crews during the development of hardware and procedures for lunar surface exploration, and oversaw final preparation of the Apollo 11 Lunar Module Descent Stage. He was designated Mission Scientist in support of the Apollo 11 mission. After training as backup Lunar Module Pilot for Apollo 15, Schmitt served as Lunar Module Pilot for Apollo 17 - the last Apollo mission to the moon. On December 11, 1972, he landed in the Valley of Taurus-Littrow as the only scientist and the last of 12 men to step on the Moon.

In 1975, after two years managing NASA's Energy Program Office, Schmitt fulfilled a long-standing personal commitment by entering politics. Elected in 1976, he served a six year term in the U.S. Senate beginning in 1977. Senator Schmitt, the only "natural scientist" in the Senate since Thomas Jefferson was Vice-President of the United States, was a member of the Senate Commerce, Banking, Appropriations, Intelligence, and Ethics Committees. In his last two years in the Senate, Schmitt held the position of Chairman of the Commerce Subcommittee on Science, Technology, and Space and of the Appropriations Subcommittee on Labor, Health and Human Services, and Education. He later served on the President's Foreign Intelligence Advisory Board, the President's Commission on Ethics Law Reform, the Army Science Board, as Co-Chairman of the International Observer Group for the 1992 Romanian elections, and as Vice Chairman of the U.S. delegation to the 1992 World Administrative Radio Conference in Spain.

Harrison Schmitt consults, speaks, and writes on policy issues of the future, the science of the Moon and Planets, and the American Southwest. He is presently Chair Emeritus of The Annapolis Center (risk assessment) and Adjunct Professor of Engineering at the University of Wisconsin-Madison, teaching "Resources from Space." Schmitt's current board memberships include Orbital Sciences Corporation, Edenspace Systems Corporation, and PhDx Systems, Inc., and, as a retired Director, he is a Member of the Corporation of the Draper Laboratory. He is a member of the Energy Department's Laboratory Operations Board. Schmitt founded and is Chairman of Interlune-Intermars Initiative, Inc., advancing the private sector's acquisition of lunar resources and Helium-3 fusion power and clinical use of medical isotopes produced by fusion-related processes.

Schmitt's honors include 1973 Arthur S. Fleming Award, 1973 Distinguished Graduate of Caltech, 1973 Caltech Sherman Fairchild Scholar, NASA Distinguished Service Award, Fellow of the AIAA, Honorary Member of the Norwegian Geographical Society and Geological Association of Canada, 1989 Lovelace Award (space biomedicine), 1989 G.K. Gilbert Award (planetology), and Honorary Fellow of the Geological Society of America, American Institute of Mining, and Geological Society of London. Dr. Schmitt has received honorary degrees from U.S. and Canadian universities. In recognition of past service, the U.S. Department of State in July 2003 established the Harrison H. Schmitt Leadership Award for U.S. Fulbright Fellowship awardees.

US MFE National Picture

N. Anne Davies
Associate Director for Fusion Energy Sciences
Office of Science
U.S. Department of Energy

Areas of Expertise
Plasma Physics
Solid state plasma physics; thin film technology

Education
Ph.D., Engineering and Applied Science, Yale University, 1972
M.Phil., Engineering and Applied Science, Yale University, 1969
M.S., Physics, Yale University, 1966
B.A., Physics, Vassar College, 1965

Professional Experience
Associate Director for Fusion Energy Sciences, 1991 - Present
Acting Associate Director for Fusion Energy, 1989 - 1991
Deputy Associate Director for Fusion Energy, 1985-1989
Director, Toroidal Confinement Systems Division, 1980-1985
Chief, Tokamak Systems Branch, 1975-1980
Physicist, Tokamak Systems Branch, 1974-1975
Research Associate, Department of Physics, University of Texas, 1972-1974

After completing her doctorate in experimental plasma physics, Dr. Davies worked in the Center for Plasma Physics at the University of Texas, developing a CO_2 laser scattering system to be used as a diagnostic to study collisionless shock waves. From 1974 on, Dr. Davies has held a series of positions in the Office of Fusion Energy Sciences, with increasing levels of responsibility.

Professional Memberships
American Physical Society
Senior Executive Association
Federal Executive Alumni Association

Awards
Fellow, American Physical Society (2003)
Meritorious Presidential Executive Rank Award (1999)
DOE Secretary's Gold Medal Award (1997)
Meritorious Presidential Executive Rank Award (1991)
DOE's Meritorious Service Award (1984)

Status of the US Inertial Confinement Fusion Program

Dr. Christopher J. Keane

National Nuclear Security Administration (NNSA), chris.keane@nnsa.doe.gov

The NNSA Inertial Confinement Fusion (ICF) Program executes the high energy density physics experiments required to support the nation's nuclear stockpile. The major facilities in the ICF Program are the National Ignition Facility (NIF), under construction at LLNL, the OMEGA laser at the University of Rochester, and the Z facility at Sandia National Laboratories. Demonstration of laboratory ignition is a major goal of the program and a necessary step for inertial fusion energy; first ignition experiments at NIF are now scheduled for 2010. The ICF Program also pursues research in other areas such as pulsed power fusion and the applications of high-energy, short-pulse lasers. Recent progress in these and other areas and a summary of future plans will be presented.

Overview of Recent Japanese Activities in Fusion Technology

M. Seki[1], I. Yamamoto[2], O. Motojima[3]

[1]JAERI, Naki, Ibaraki, Japan,[2]Nagoya University, Nagoya, Japan,[3]NIFS, Toki, Japan

After ITER/EDA, Japanese activities in fusion technology have been mostly devoted to R&Ds towards DEMO reactors. The paper intend to overview these activities.

With respect to the test blanket module development, solid breeder blankets with ferritic steel structure cooled by helium and water are being developed by JAERI in cooperation with universities and NIFS. While solid breeder blankets with SiC composite structure cooled by high temperature helium gas, liquid LiPb breeder cooled by helium, molten salt self cooled blankets and liquid Li self cooled blankets are under development by universities and NIFS in cooperation with JAERI.

In terms of the superconducting magnet development, JAERI has performed a basic research for the Fusion Power Demonstration Plant, aiming at realization of toroidal filed higher than 13 T. Innovative superconductors, such as Nb_3Al and High Temperature Superconductor (HTS), are candidates for this application. As an intermediate step, 1.5-m diameter coil using Nb_3Al conductor has been built and successfully operated at 46 kA in the background field of 13 T, which is the first demonstration of Nb_3Al applied to a large-scale magnet. A 60-kA HTS current lead has also been developed.

In the R&D of negative ion based NBI technologies, a H^- beam of 110 mA has been stably accelerated up to 0.9 MeV, which corresponds to the current density of 80 A/m^2. A long pulse beam injection of 17 s has been performed with 1.6 MW at 360 keV from one ion source in the JT-60 N-NBI. A beam power of 13.1 MW at 180 keV has been injected from three injectors in the LHD N-NBI.

In the radio-frequency heating technology, development of 170GHz ITER gyrotron has been progressed to achieve a 500kW for 100 sec operation in JAERI RF test stand. In JT-60 experiment, 2.8 MW power was injected into plasma for 2.8 seconds at 110 GHz. Long pulse injection for 756 sec with 72 kW was achieved in LHD ECH experiment at 84 GHz. In the test of a remote steering antenna for ECH, efficient transmission of 95% was successfully obtained at high power of 500kW using JAERI 170GHz gyrotron RF source. In the ICRF, a new record of long pulse up to 150 sec was achieved with 500 kW injection into LHD plasma.

In the area of tritium processing technology, R&D has been focused on the blanket tritium recovery technology. Tests on interaction between a tritium recovery system and a fuel cycle system has started using an integrated model system. Development of advanced techniques for tritium removal or detritiation has also progressed together with the reliability confirmation study of present detritiation system.

Oral Session I-1

Engineering of Experimental Devices

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Advanced Control Techniques and High Performance Discharges on DIII-D[*]

A.G. Kellman for the DIII-D Team

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The goal of the DIII-D advanced tokamak program is to provide the scientific basis for the optimization of the tokamak approach to fusion energy. A critical element in achieving that goal is the development of new control techniques for regulation of current and pressure profiles and suppression of plasma instabilities that enable plasma performance to be improved over the conventional tokamak. The DIII-D program has made significant progress in developing and implementing these advanced control techniques and has successfully achieved enhanced operating modes that have a strong impact on the technology for future fusion devices.

We continue to expand our electron cyclotron heating (ECH) and current drive (ECCD) system and now have a 4 MW, 5 gyrotron system with fully steerable launchers. We have developed advanced real-time control algorithms to locate and align ECCD deposition with magnetic islands to suppress both the 3/2 and 2/1 neoclassical tearing modes. Real-time calculation of the internal magnetic flux surfaces has allowed the ECCD/island alignment and mode suppression to be maintained while the plasma energy is further increased. The addition of a 12-element internal non-axisymmetric control coil system, the I-coil, has expanded our ability to stabilize the resistive wall mode (RWM), an instability that limits plasma performance at high plasma pressure. In a unique application of the I-coil coupled with the previous six-element external coil, an edge ergodic field has been produced that significantly reduces the occurrence of the large edge localized modes (ELMs) and their resultant high pulsed heat loads. Recent upgrades to our digital plasma control system speed enable more extensive implementation of advanced control algorithms including real-time calculation of the safety factor profile, q(r), that is needed for current profile control.

Using these and other control techniques, we have produced a variety of high performance discharges that represent significant progress toward successful realization of advanced tokamaks. Improved stabilization of the RWM has permitted robust operation up to β_N ~3.8, well above the conventional tokamak no-wall beta limit. By combining ECCD, neutral beam injection and high bootstrap current, we have demonstrated 100% noninductive current drive in discharges above the no-wall beta limit ($\beta_N \sim 3.5$). Using more conventional control techniques, we have achieved a hybrid mode of operation applicable to ITER that would enable operation at β_N ~3 at q_{95}=4.4 and Q=10 for a duration of 4500 s.

The DIII-D program is planning an aggressive set of system upgrades in 2005–06 that will further enhance our capabilities to investigate advanced tokamaks: expansion of the EC system to 6 MW, 10 s will permit improved current profile control; a new lower divertor baffle will permit pumping and more efficient current drive in highly triangular double-null discharges; and reversal of a neutral beamline will permit investigation of the ELM-free H–mode regime and RWM stabilization in low rotation discharges.

[*]This work was supported by the US Department of Energy under Cooperative Agreement DE-FC02-04ER54698.

W7-X Progress

M. Gasparotto and the W7-X team

Max-Planck-Institut für Plasmaphysik, Euratom Association, Teilinstitut Greifswald, Wendelsteinstraße 1, D-179491 Greifswald, Germany

The WENDELSTEIN 7-X stellarator (W7-X) is the next step device in the stellarator line of IPP and is presently under construction at the Greifswald branch institute. The experiment aims at demonstrating the steady state capability of a stellarator machine at reactor relevant parameters.

The main parameters of W7-X are: average major radius 5.5 m, average plasma radius 0.53 m, magnetic field on the plasma axis 3.0 T, total weight 725 t. An important feature of W7-X is the high geometrical accuracy of the magnetic field configuration; perturbations ($\Delta B/B$) with a periodicity different from the five-fold periodicity of the device should be kept below 10^{-4}. The magnetic system of the machine consists of 50 non planar and 20 planar superconducting coils supported by a central structure made of 10 sectors. The magnetic system is kept at 4K by liquid helium.

The superconducting coils are made by a conductor composed of 243 NbTi/Cu strands twisted to form a cable that is enclosed in an aluminium alloy jacket. The winding package is insulated by fibre glass epoxy resin and enclosed inside a stainless steel casing. To limit the ohmic heat production, electrical joints between superconductor cables with an electrical resistance below 1 nΩ at 4 K, were successfully developed. Conventional current leads appropriately designed are used to connect the seven groups of superconducting coils with the power supplies (20 kA-30 V each).

Critical components are the coil support elements connecting the coils to the central structure and the inter-coil support elements connecting the coils one to the other. These supports operate in high vacuum and at cryogenic temperature, withstand high loads and moments and allow the assembly of the machine with high accuracy.

Efficient thermal insulation of the superconducting coils is achieved by high vacuum and multilayer insulation. The complex shape of these thermal shields need the development of support panels based on a new technology: glass fibre epoxy resin reinforced with integrated Cu-mesh.

The plasma vessel is composed of 10 half-modules and is being constructed from stainless steel rings bent precisely to the shape and successively welded together to keep the vessel surface within tight tolerances.

The high thermal load from the plasma (up to 10 MW/m^2) is taken by a divertor made by Carbon Fibre Composite tiles brazed on water cooled CuCrZr alloy heat sinks.

A 10 MW ECRH system with CW-capability operation at 140 GHz is under construction to meet the scientific objective of W7-X. The microwave power is generated by 10 gyrotrons; a prototype was successfully developed and tested. The microwave power is focussed into the plasma through 10 synthetic diamond barrier windows and quasi-optical plug-in launchers.

Production of most of the W7-X components made significant progress: some planar and non planar coils have successfully tested, segments of plasma vessel were delivered at Greifswald and most of the tools required for the assembly are ready for use.

The paper will report the recent progress on W7-X with particular emphasis on the components where high-technology solutions have been applied.

Recent Progress of Low Aspect Ratio Machines

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Research on the properties of the Spherical Torus for development of fusion science and technology encompasses a broad effort across several experimental devices. This proof-of-principle program is focused on studying high-temperature plasmas at very low aspect ratios. The two largest experiments, NSTX and MAST, are addressing critical physics issues at the $I_p \sim 1$ MA level. This allows exploration of the potential of the concept for a cost-effective contribution to fusion energy development both as a fusion concept in its own right and as a means of advancing key tokamak physics concepts at geometric extrema.

Smaller experiments in the US are focused on specific ST-related and/or fusion issues. The HIT program is developing and testing the coaxial helicity injection approach for startup and current drive. CDX-U is testing Li-liquid limiters and progressing to a full Li-wall tokamak test. The PEGASUS program is exploring the extreme near-unity aspect ratio regime to establish the limits of ST parameter space.

High beta plasmas are readily achieved in ST experiments, with $\beta_N > 5$ being achieved through increasing plasma control and improved confinement with H-mode operation. While tokamak-like confinement is achieved, the absolute values are seen to be higher than standard ITER scaling.

Some critical issues for future development include noninductive plasma startup, sustainment, and energy exhaust. Electron Bernstein Wave heating and current drive is being actively pursued in both the largest and some smaller experiments. Helcity injection, neutral beam current drive, and RF drive schemes are all under investigation for future application. Studies of H-mode and divertor operation in STs are revealing new insights into power flows during ELMs and ELM structure itself.

These experiments are generating relevant experience with the technical challenges of constructing and operating tokamak-like devices at near unity aspect ratio. The large electromechanical stresses in the centerstack region, plus the lack of space in that region, require both new engineering and scientific approaches to generating and sustaining the plasma.

Based on the progress to date, initial considerations of the design of a performance-extension ST experiment are underway in the U.S. A device of a scale comparable to that envisioned for a component test facility would require R ~ 1.5 m, Ip ~ 10 MA, and pulse lengths of 5 – 50 sec to provide the required physics basis, and could be considered in parallel with larger efforts in the international fusion program.

Progress in Technology at JET

A S Kaye[1], P Coad[1], T C Jones[1], A Lioure[2], J Pamela[2], A Rolfe[3], S Rosanvallon[2], T Todd[1]

[1] Euratom UKAEA Fusion Association, Abingdon, OX14 3DB, UK; ask@jet.uk
[2] EFDA CSU, Abingdon, OX14 3DB, UK
[3] Oxford Technologies Ltd, Abingdon, OX14 3DB, UK

JET has contributed strongly to the development of fusion technology over many years, and technical development continues in parallel with the operation of the Tokamak. These programmes are supported by the associated laboratories around Europe and elsewhere, including US support for a number of activities. A major shutdown extended through much of 2001 and the machine is currently shutdown for 11 months for the installation of many further enhancements.

JET operations have included both the study of ITER-like scenarios and of technical objectives in support of ITER, notably mitigation of ELMs and disruptions, investigation of tritium retention, development of burning plasma diagnostics, real time control of plasma parameters, and control of extreme plasma shapes to centimetre tolerance. Safe operation of high force ITER scenarios with disruptions up to 500 tonnes vertical force, and operation at 4 Tesla toroidal field, have been demonstrated in recent campaigns with low consumption of the fatigue life of the machine, less than 10% over the lifetime of JET. JET heating systems are being upgraded, with additional new NB power supplies and uprated ion sources having recently been brought into full operation, and an ITER-like ICRF antenna under manufacture. Coupling of LHCD over large distances to the separatrix has been demonstrated. Other enhancements have included installation of four external error field correction coils, and a quartz microbalance mounted in the divertor to monitor carbon film deposition.

During the present shutdown, around 20 new or improved systems are planned to be installed. These are mainly diagnostics, including various tritium retention diagnostics, magnetic and halo current probes, neutron and alpha particle detectors, and TAE antennae. Modifications to the divertor will enable increased triangularity ITER-like scenarios. The bulk of the in-vessel installation will be carried out using the extensive JET remote handling equipment to minimise the cumulative dose. Important up-grades of the RH tools have been implemented, notably the provision of force feedback to enable the remote installation of loads up to 260kg, and the extensive use of virtual reality methods for preparing and implementing the work.

Analysis of tritiated samples from the tritium campaign in 1997, including highly tritiated flakes from near the divertor, continues as does the development of methods for detritiation of hard and soft waste and of tritiated water. The unique JET capability to operate with tritium (and beryllium) has been maintained; a further trace tritium campaign was completed in autumn 2003 during which 5 grams of tritium were supplied to the machine, mainly to one of the NB injectors. The experience in the use of tritium over an extended period, and in implementing in-vessel enhancements on an active and tritiated machine, is a valuable input to the preparations for tritium burning Tokamaks. Notwithstanding the on-going use of tritium, it remains practical to implement important enhancements of JET to fully exploit its capabilities in the preparations for ITER operation.

Recent Progress in the Design, R&D, and Fabrication of NCSX

R.T. Simmons[1] on behalf of the NCSX Team

[1]*Princeton Plasma Physics Laboratory, Princeton, NJ, reiersen@pppl.gov*

The NCSX project is an innovative magnetic fusion configuration consisting of a stellarator plasma with three field periods surrounded by eighteen modular coils (six per field period). A vacuum vessel fills the internal volume of the modular coils to provide maximum space for plasma shape flexibility. The complete NCSX system also includes toroidal (TF) and poloidal field (PF) coils, a cryostat, power subsystems, and numerous other subsystems.

The design of the stellarator core is proceeding well. Considerable analysis and R&D have been performed to underpin the design. Key analyses include structural analysis of the modular coils, TF and PF coils, and vacuum vessel. Prototype modular coil winding forms and prototype vacuum vessel segments have been fabricated to establish the manufacturability of the designs and to qualify suppliers for the production contracts. Extensive winding R&D has been conducted, including the fabrication of several prototype articles with full scale cross-sections. Testing to determine the properties of the epoxy impregnated cable conductor has been conducted. Analysis and R&D results are presented and discussed.

NCSX fabrication activities have begun. A winding facility for the modular coils has been constructed including clean rooms, turning fixtures, and an autoclave. A test facility for testing the modular coils at cryogenic temperatures has also been constructed. The NCSX Test Cell has been cleared of legacy equipment and the shield walls are being reconfigured. Long lead procurements for the modular coil winding forms and vacuum vessel subassemblies are being placed with industry. NCSX fabrication activities are discussed.

Procurement plans and plans for field period assembly, final assembly, and integrated systems testing are in place. Future plans for getting to First Plasma are presented.

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Oral Session I-2

High Average Power Laser - Special Session

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The Path to Develop Laser Fusion Energy*

J. D. Sethian[1] and S.P. Obenschain[1]

[1]*Plasma Physics Division, Naval Research Laboratory, Washington DC 20375*
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We are developing the science and technology for fusion energy with laser direct drive targets and solid wall chambers. The modular nature of this approach and the inherent engineering advantages reduces both the cost and risk of development. These factors, plus the recent scientific and technical advances, make laser fusion an increasingly attractive path to a practical fusion energy source. This talk will present the program development philosophy, an overview of the technical progress, and the path to laser fusion energy.

Technology development is carried out through The High Average Power Laser (HAPL) program, a multi-institutional, multi-disciplinary effort that follows two core beliefs: 1) The research should be guided by the end goal of a practical power plant. 2) The main components should be developed in concert with one another to ensure they are developed as an integrated system. These components are: the lasers [krypton-fluoride (KrF) and diode-pumped solid-state laser], chamber, final optics, target fabrication and target injection. This work is coordinated with the direct drive target designs carried out largely through the DOE/ICF program.

Significant advances have been made in all major areas: High-resolution 2-D simulations of target designs predict gains greater than 150, which is sufficient for net power generation with laser fusion. Both types of lasers now run repetitively pulsed and have the potential for meeting the durability and efficiency requirements. Grazing incidence aluminum mirrors have been shown to meet the needed reflectivity and laser damage threshold. A target injector has been brought to operation. The foam shells needed for the targets have been made in batch quantities with the proper dimensions. Experiments show DT ice grown over a foam underlay both exceeds the smoothness requirements and is thermally robust. And finally, progress has been made in the materials needed for the chamber first wall. Other talks in this HAPL session will discuss some of these advances in detail: the lasers, the first wall, and a chamber dynamics model to study how fast the chamber clears itself between shots. Other presentations at this conference will discuss target fabrication, target injection, and materials development.

We propose to develop and demonstrate laser fusion energy in three phases. The present Phase I program includes HAPL and is developing the critical science and technologies. Phase II would develop and integrate full size components. Phase III, the Engineering Test Facility (ETF), would have three functions: 1) optimize laser-target and target-chamber interactions, 2) develop materials and components; and 3) generate net electricity fusion. We could be technically ready to start construction of the ETF within the next decade and start operations by 2020. This development could allow construction of pilot commercial plants well before 2050.

Work sponsored by US Department of Energy, NNSA

*This is a summary of work performed by researchers from over 20 institutions. For a detailed list of collaborators, see: J.D. Sethian, et al, “Fusion Energy Research with Lasers, Direct Drive Targets, And Dry Wall Chambers,” Nuclear Fusion, **43**, 1693-1709 (2003).

KrF Laser Drivers for Inertial Fusion Energy

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The Electra laser facility at the Naval Research Laboratory (NRL) is developing the science and technology for a repetitively pulsed, electron beam pumped, krypton fluoride (KrF) laser for inertial fusion energy (IFE). KrF lasers are ideally suited for IFE due to their established uniform beam quality, short lasing wavelength (248 nm), and pulsed power architecture that scales to large systems. The technologies developed on Electra must meet the fusion energy requisites for repetition rate, efficiency, durability and cost. Although Electra is approximately 1-2% of the energy of a power plant size laser beam line, it is large enough that the components that are developed are directly scalable to a full-sized IFE facility. Specifically, the KrF laser design includes: a durable, efficient, and cost effective pulsed power system; a durable, uniform cathode (electron beam source); a long life, e-beam transparent pressure foil structure; a gas recirculator to cool and quiet the laser gas between shots; and long life, fluorine resistant optical windows and coatings.

The Electra laser is pumped with two V = 500 kV, I = 110 kA, t = 140ns (FWHM) electron beams. The beams are injected into a laser cavity volume that measures 30x30x100 cm^3 with the laser aperture along the long axis. To date, Electra has produced an output energy of 700 J per pulse at a repetition rate of 1 Hz and 625 J per pulse at 5 Hz, with efficiencies that scale to > 7%, which is within the IFE requirement. The laser performance has been modeled with the "ORESTES" KrF physics code developed at NRL. Recent experiments show long-lived pressure foils can be achieved by periodically deflecting the laser gas to cool the foils. This paper discusses an overview of the objectives of Electra KrF laser program; a background of the physics and technology of KrF lasers; the most recent advances in laser performance, KrF physics modeling, system efficiency, cathode development, and foil lifetime; and the technical challenges that still need to be addressed.

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Diode-Pumped Solid-State Laser Driver for Inertial Fusion Energy

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Flashlamp-pumped Nd:glass lasers have served as the workhorse driver technology to explore the physics of inertial confinement fusion (ICF), owing to their scalability (>10 kJ/beamline) and flexibility in pulse-shaping. Frequency-conversion to the ultraviolet and spectral bandwidth have been accommodated as well. KrF fusion lasers are also a viable driver option with certain advantages, such as very smooth beams. This paper is confined to a discussion of diode-pumped solid-state lasers (DPSSLs), and their ability to address the requirements of ICF and inertial fusion energy (IFE).

Our DPSSL design for IFE is optimized with two amplifier heads arranged in a four-pass architecture known as the Mercury laser. We have previously activated a single amplifier and achieved an energy of 34 J/pulse single-shot, whilst 23 J/pulse has been demonstrated at 5 Hz operation for ~10^4 shots. These results have validated the functionalities of the very large diode arrays, gas-cooling of the Yb:S-FAP laser crystals, and the laser architecture. Although the risk level is significantly mitigated with this demonstration, additional challenges are needed to activate the second amplifier head. In particular, the Yb:S-FAP slabs must operate at higher damage threshold, and the beam modulation at the "reverser" mirror must be reduced. Our main strategy has been to employ a state-of-the-art polishing method known as magneto-rheological finishing to reach our goals, in addition to implementing several optical design improvements. Activation of the second amplifier is expected to commence within a few months, and last for 6 months (while energy levels are raised from 50 J to 100 J).

Our analysis of a full-aperture beamline suggests that 4 kJ should be attainable. We have been analyzing this system, paying careful attention to the fundamental physics issues that may limit its performance, such as stimulate Raman scattering, nonlinear growth, and amplified spontaneous emission. Solid-state laser drivers require frequency conversion and beam smoothing, and our near- and longer-term plans in this area will be discussed. The current status and future of solid-state lasers will be addressed with respect to the efficiency, smoothness, reliability, and repetition-rate requirements of IFE.

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Effects of Chamber Geometry and Gas Properties on Hydrodynamic Evolution of IFE Chambers

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Understanding the long-timescale, hydrodynamic evolution of IFE chambers following a target explosion is essential for the design and operation of rep rated fusion facilities. Target injection and survival, propagation of laser beams, clearing of chambers, *etc.* critically depend on the chamber environment. The chamber state evolution over hundreds of milliseconds is dependent on a variety of factors, such as chamber geometry, chamber constituents (*e.g.,* gases, aerosol) and their properties and the chamber pre-ignition conditions.

SPARTAN simulation code has been utilized to investigate the IFE chamber evolution. SPARTAN solves full Navier-Stokes equations in 2D Cartesian and cylindrical coordinate system and can handle arbitrary-shape boundaries. The uniform accuracy of the solution on the entire grid domain is achieved by adaptive mesh refinement. We have simulated the dynamic evolution of a 6.5-m-radius chamber filled with either Xe, D, or He at different initial densities. Xenon has been proposed previously as a protective gas. Following the target explosion D, T, and He ions from the target will be implanted in the wall and will eventually diffuse back into the chamber. Initial conditions for SPARTAN are taken from solutions of BUCKY 1-D rad-hydro code. Sutherland law is used to extrapolate thermal conductivity and viscosity data to higher gas temperatures and ideal gas law is used as the equation of state. Chamber wall is assumed to have a constant temperature of 700^{o}C.

The results indicate that thermal conductivity and viscosity of chamber gas as well as the 2D geometrical effects have a major impact on the evolution of the chamber environment. For example, the size and distribution of eddies in the chamber is dependent on the gas viscosity and on the number and configuration of the laser beam channels. The cooling mechanism is driven by heat conduction from gas into the wall and by turbulent mixing combined with thermal diffusion. Shock waves generated from the heating of the chamber gas by the blast are reflected from the chamber wall and produce a host core by compressive heating. For the case of Xe, this hot core can reach a temperature of 20-40 eV. As such, significant ionization exists and radiation may cool this region. The impact of this background plasma on the evolution of the chamber is under study.

Development of a Dry Wall Concept for Laser IFE Chambers

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The first wall of a laser fusion chamber will experience high heat loads pulsed at 5-10 Hz with pulse widths on the order of a few microseconds. This poses a challenging problem for dry wall designs, as the wall will experience high stresses and will thus be susceptible to a variety of failure modes. The primary design concept of the High Average Power Laser (HAPL) project is a ferritic steel first wall coated with tungsten armor. Due to the high heat loads, the armor will experience high temperatures (~2500 C), extensive yielding, and surface cracking. In order to evaluate the ability of this design to provide a suitable lifetime, a series of experiments to simulate chamber conditions using ions, x-rays, infrared heating, and lasers is planned. These experimental efforts will be coupled with numerical modeling to help determine likely failure modes and establish design criteria for chambers.

The modeling is carried out using a commercial finite element code (ANSYS) to analyze the temperatures and stresses in the test coupons for each of the experiments. The heating from the various sources is included as volumetric heating, with the depth profile of the heating depending on the spectra of the incident energy and the stopping power within the tungsten. Temperature dependent properties are used throughout, and bilinear stress-strain models are used for the tungsten. Fatigue analysis is used to estimate the number of cycles needed for cracks to develop at the tungsten surface, and more detailed fracture mechanics analysis, using path integrals, is used to determine the depth to which these cracks are likely to propagate. These models are used to correlate the thermomechanical effects in the samples by comparing the results of the models with those of the post-test characterization. The samples are characterized using surface roughness and crack morphology as the primary characteristics of interest. From these comparisons, we elucidate the failure mechanisms of the steel/tungsten wall and attempt to infer the lifetime of such a wall.

The x-ray, ion, and laser experiments are designed to simulate the surface temperature expected in the HAPL chamber. This will help us to better understand the surface roughening and fracture seen in earlier experiments. The infrared experiments do not have sufficient power to mimic the surface temperatures of HAPL, but they do have sufficient time-averaged power to simulate the thermal and stress histories near the tungsten/steel interface. Hence, this experiment will be useful for addressing such issues as delamination and cracks penetrating the tungsten and reaching the steel. Again, modeling will be used to correlate these results with predictions, allowing us to infer failure mechanisms.

Finally, we also present a model for a larger, cooled component tested with the infrared source. This larger component will allow us to address failure mechanisms that will not be present in coupon tests, but are expected in reactor applications.

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Oral Session I-3

Socioeconomics, Safety, Radwaste, and Licensing

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Future World Energy Demand and Supply: China and India and the Potential Role of Fusion Energy*

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Massive increases in energy demand are projected for countries such as China and India over this century, e.g., many 100s of MWe of additional electrical capacity by 2050, with more additions later, are being considered for each of them. All energy resources will be required to meet such a demand. Fortunately, while world energy demand will be increasing, the world is well endowed with a variety of energy resources. However, their distribution does not match the areas of demand and there are many environmental issues. Such geopolitical issues affect China and India and make it important for them to be able to deploy improved technologies. International collaborations in developing these technologies, such as ITER, may be important in all energy areas. In this regard, Korea is an interesting example of a country that has developed the capability to do advanced technologies - such as nuclear power plants. Fusion energy is viewed as an interesting potential option in these three countries.

* This presentation is based upon a workshop held at IPP in Garching, Germany on December 10-12, 2003.

Status of Safety and Environmental Activities in the US Fusion Program

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The safety and environmental (S&E) advantages of fusion have been recognized since the earliest days of the US fusion program. Over the past 25 years, the magnetic fusion energy (MFE) Fusion Safety Program at the Idaho National Engineering and Environmental Laboratory and the inertial fusion energy (IFE) safety group of the Fusion Energy Program at Lawrence Livermore National Laboratory have been conducting safety research and development (R&D) and S&E assessments of conceptual designs to demonstrate the S&E potential of fusion.

S&E research is focused on understanding the behavior of the largest sources of radioactive and hazardous materials in a fusion facility, understanding how energy sources in a fusion facility could mobilize those materials, developing integrated state of the art S&E computer codes and risk tools for safety assessment, and evaluating S&E issues associated with emerging design concepts in the fusion community. Some of the largest sources of hazards are chamber material reactions with air or steam, and chemical toxicity of fusion materials. A collaborative survey of the S&E characteristics - both radiological and chemical - of various IFE candidate target and coolant materials has been completed. Mobilization studies have addressed tungsten and Flibe mobilization, as well as other materials. Integrated state-of-the-art S&E analysis tools have been developed to address a variety of MFE and IFE safety issues, including magnet arcing, thermal hydraulic/source terms, tritium migration, and neutron transport and material activation, for accident analysis and radioactive waste assessment. Some of these tools have also supported Inertial Confinement Fusion and accelerator facilities, such as the National Ignition Facility and the Rare Isotope Accelerator, respectively. Waste management issues continue to be investigated and are receiving increased attention. Risk tools for safety assessment have produced a failure rate database, augmented a radiological dose code for fusion use, and developed sets of accident initiators for MFE and IFE. Our evaluations of S&E issues associated with emerging IFE design concepts include support to the High Average Power Laser (HAPL) program to advance the science and technology for a dry-wall, laser-driven IFE power plant, and collaboration with the Heavy Ion Fusion (HIF) and Z-Pinch programs for the development of alternative, thick-liquid-wall IFE concepts. Recent S&E support to MFE design concepts includes APEX and ARIES, and burning plasma experiment designs, including preliminary studies of the Fusion Ignition Research Experiment (FIRE) and detailed accident analyses for the International Thermonuclear Experimental Reactor (ITER).

Excellent progress has been made in understanding the nature of the S&E concerns associated with magnetic and inertial fusion. This paper presents key R&D highlights over the past 15 years, reviews recent safety assessment results for both MFE (e.g., APEX, ARIES, FIRE, ITER) and IFE (e.g., HYLIFE-II, SOMBRERO) designs, and discusses impact of the results on future programmatic directions in the fusion program.

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Evolution of Clearance Standards and Implications for Radwaste Management of Fusion Power Plants

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The issue of radioactive waste management presents a top challenge for the nuclear industry. As an alternative to recycling or disposal in repositories, many countries are proceeding successfully with the process of developing clearance guidelines that allow solids containing traces of radioactive materials to be cleared from regulatory control and unconditionally released to the commercial market after a 100 year storage period. While the external components surrounding the nuclear island could meet the clearance requirements, researchers have constantly applied the clearance criteria to the in-vessel components as well in an attempt to further minimize the volume of waste assigned for geological burial in repositories.

The clearance limits developed by the International Atomic Energy Agency (IAEA) over the past decade have been used worldwide for diverse range of fusion concepts from MFE tokamaks and stellarators to IFE laser, heavy ion, and Z-pinch applications. With the emergence of the new 1996 European Union (EU) clearance standards by EURATOM and the more recent US guidelines for solid materials by the Nuclear Regulatory Commission (NRC), we took the initiative to compare the IAEA, EU, and US-NRC clearance limits in order to identify the implications on the ARIES fusion waste management approaches and highlight the areas of discrepancy and agreement for the isotopes of interest to fusion applications. For this purpose, we employed a simplified model in which SiC-based and FS-based systems undertook the appropriate arrangement of the in-vessel and ex-vessel components using the physical and operating parameters of the ARIES-CS compact stellarator power plant.

We observed a notable difference between the clearance limits for the 1650, 300, and 67 radionuclides developed by the IAEA, EU, and US-NRC, respectively. At first glance, we noticed that the US-NRC standards are the most conservative, followed by the IAEA's, then the EU's. However, applying the limits to the ARIES-CS design, the trend was reversed with the EU clearance index being the highest for all components at 100 y after shutdown. According to the three standards, none of the power core components (blanket, shield, vacuum vessel, and magnet) can be cleared after the 100 y storage period as their clearance indexes exceed unity by a wide margin. The building that surrounds the power core is subject to a less severe radiation environment and thus contains residual radioactivity. The building represents no risk to the public health and safety and it appears feasible to release its constituents (concrete and reinforcing mild-steel) to the commercial market or nuclear industry after a relatively short storage period of 25 y or less, depending on the limit. Of interest is that the building dominates the low-level waste stream and its release saves a substantial disposal cost for such a large quantity, freeing ample space in the repositories for higher level wastes.

This exercise is proving valuable in understanding the differences between the various clearance standards. While US clearance standards now exist for a limited number of radionuclides that are important to the fission industry, no such standards are in place for radionuclides of interest to fusion facilities. Before fusion penetrates the energy market, the US-NRC should develop fusion-specific standards that address the safe release of fusion solids with trace levels of radioactive materials.

Key Issues for the Safety and Licensing of Fusion

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The safety and environmental characteristics of fusion as a power generating source provide important motivations for its development. The low stored energies, benign reaction products, absence of climate-changing emissions, and rapid termination of power excursions are amongst the features that naturally give fusion a safety advantage. But in order to license the construction and operation of a fusion power plant, a high degree of assurance of these benefits may be required. Although the regulatory regime in which future power stations will be approved is not well known, some requirements can be anticipated, particularly with the benefit of experience so far in preparations for the licensing of ITER.

Regulatory requirements for fusion power plant, as for any other large industrial installation, are likely to focus on hazards to the public and to workers at the plant, and on longer-term impacts on the environment. The extent to which these hazards can be reduced or avoided by inherent or passive features will be an important measure of attractiveness.

Over the past ten years, various studies of conceptual fusion power plant designs, as well as the engineering design of ITER, have included comprehensive analyses of safety and environmental performance. These have indicated that the accident potential in a fusion plant is indeed very low: the likelihood of an event leading to a public hazard is very low and the potential consequences are also very low. Occupational safety, while on a level comparable with other power-generation technologies, is an area where some improvement may be achieved by optimization of design with regard to maintenance requirements. This in turn depends on materials development to extend the lifetime of components. Operational releases to the environment from a fusion plant should be negligible, however at the end of life there remains a large volume of material that has been activated by neutrons. Most of this is at a low level of activation that decays rapidly, so that it no longer presents a radiological hazard and could be cleared from regulatory control. But full use of materials recycling may be necessary to avoid a small fraction of the total requiring disposal as long-term waste.

Each of these areas will be reviewed, with reference to previous studies of ITER and of power plants, in particular the recently completed Power Plant Conceptual Study in Europe. In the context of anticipated regulatory requirements, the key issues will be highlighted and discussed.

Fusion Power; a Strategic Choice for the Future Energy Provision
Why is so Much Time Wasted for Decision Making?

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From a general analysis of the world energy issue, driven by a variety of elements such as the enhanced greenhouse effect, the strong fluctuations in energy prices, the world-wide drive towards liberalization of the grid-based energy carriers, and the social-driven tendency to integration of renewable energy sources and decentralized generation, it is argued that an affordable, clean and reliable energy supply will have to consist of a portfolio of primary energy sources, a large fraction of which will be converted to a secondary carrier in large base-load plants. Because of all future uncertainties, it would be irresponsible not to include thermonuclear fusion as one of the future possibilities for electricity generation.

Given this conclusion, the author tries to understand why nuclear-fusion research does not really seem to be considered as being of strategic importance by the major world powers. The fusion programs of the USA and Europe are taken as prime examples to illustrate the 'hesitation' of these economic giants. Recently, Europe has started to advocate a so-called 'fast-track' approach, thereby abandoning the 'classic' route towards fusion that it has projected for many years. The US 'oscillatory' attitude towards ITER in relation to its domestic program is a second case study that is looked at.

However, from the real history of the ITER design and the 'siting' issue, one can try to understand as to how important fusion is considered by these world powers. Not words are important, but deeds. Fast tracks are nice to talk about, but timely decisions need to be taken and sufficient money is to be provided. More fundamental understanding of fusion plasma physics is important, but in the end, real hardware devices must be constructed to move along the path of power plant implementation.

The author tries to make a balance of where fusion power research is at this moment, and where, according to his views, it should be going. Finally, the roles of Japan and Russia, are explored and their perceived approaches are compared to the US and EU fusion 'road maps'.

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Poster Session I

Code Development, Testing, and Validation Experiments

Target Development and IFE Technology

Nuclear Analysis and Experiments

In-Vessel Components and Power Conversion

Materials Development

Magnets and Structural Analysis

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In-pile Assemblies for Testing of Li_2TiO_3 Lithium Ceramic Blanket

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Objectives. Carrying out the lithium ceramics radiation trials with the ^6Li 96 % enrichment and *Li* burnup up to ~20% and "in situ" registration of released tritium under various temperatures.

Methodology. The IVG.1M and WWR-K reactors of the National Nuclear Center RK were used for irradiation. Two type ampoules – active and passive ones – were used during irradiation. "Active" ampoules are intended for study of tritium release dynamics and contain capsules with ~1 mm lithium ceramic pebbles. Capsules in ampoules A1 and A2 have temperatures, which change within the campaign in the range 400-900 °C, ampoule A3 has fixed temperature of 650 °C. "Passive" ampoule is only intended for production of samples with *Li* burnup 20%; it contains capsules with ~1 mm pebble samples, and pellets with 8 mm diameter and 1 mm thickness. During testing the temperature of capsules in ampoules P1, P2, and P3 is fixed and equals 400, 650 and 900°C respectively. Every ampoule contains 2 grams of lithium ceramics Li_2TiO_3.

During long-term tests the temperature of lithium ceramic sample is regulated by changing the gas (helium) pressure in ampoules using gas-vacuum system of universal loop facility. The automated system for mass-spectrometer registration of hydrogen isotopes is used to measure tritium released from samples during reactor irradiation.

Results. The preliminary tests were carried out at the IVG.1M reactor. The tests' goal is development and experimental examination of ampoules on the basis of thermo-physical and neutron-physical calculations, work-out of temperature regulation modes and measurement of tritium release dynamics during reactor irradiation. In accordance with work results the experimental technique for long-term irradiation of lithium ceramic during 200 days was validated.

Validation of calculation models and experimental study of space-energy distribution of neutrons and reactivity effects were carried out at the critical test bench by using ampoule assembly physical mock-up. For that the WWR-K reactor core configuration was simulated at the critical test bench. Experimental value of lithium ceramic energy release was measured in the WWR-K reactor by using calorimetric method. This value is used for assessment of lithium burnup.

At present long-term irradiation is carried out in two irradiation channels of the WWR-K with simultaneous "in situ" registration of released tritium under various temperatures. It is planned to carry out the posterior material testing study of irradiated ceramic Li_2TiO_3.

KTM Experimental Complex Project Status

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The KTM tokamak is intended for study and tests of materials and structures of first wall armor, receiving divertor plates and divertor components under heat load modes similar to ITER and future fusion reactors. The experimental complex is constructed by Kazakhstani and Russian organizations in National Nuclear Center RK, Kurchatov.

The facility's design features are movable divertor device (MDD) and transport sluice device (TSD), due to them the unique possibility is to access to vacuum chamber and to change divertor plates without high vacuum failure. The MDD serves for placement of removable components receiving plasma thermal and particle fluxes, provides for their vertical positioning, replacement of all these components by using one loading sluice. Number of removable components is 24, number of full-scale cycles of plasma discharge - 2×10^4. Number of plasma disruptions is 10^3. The sluice is designed on not less than 2000 operation cycles without of serviceability infringement.

At present the initial data and documentation have been developed for designing vacuum-technological complex, the KTM chamber heating and conditioning system and magnetic system, as well as the KTM thermo-mechanical state monitoring system. The following activities have been carried out as well: development of design documentation and initial data of KTM physical diagnostics system, KTM vacuum chamber, RF-heating system, mock-up and execution of design decisions for systems of automation, control and emergency protection. The activities on reconstruction of existing buildings, construction of new ones, as well as mounting of the KTM complex external power supply systems have been begun. The amplifying calculations of basic scenario have been carried out; the disruptions (up and down) have been defined; verification calculations of structure taking into account disruptions have been carried out.

Moreover, considering uniqueness of this facility, preliminary development of experiments and checkout of research techniques, as well as personnel training shall be required for project successful realization. With this purpose the works on creation of the following experimental test benches have been begun: *technological test bench* – for vacuum-technological preparation of discharge chamber, including the development of boronization technology, technology of differential vacuum pumping of the divertor volume, technological modes of plasma additional RF-heating system, *test bench for simulation tests* – for creation of experimental facilities modeling impact conditions (plasma fluxes, neutron irradiation, etc.) on materials of the KTM components, *test bench for physical diagnostics* – to setup and test systems of diagnostic measurements, to adjust system of synchronization and automation of measurements and data processing. The KTM facility commissioning is scheduled at the end of 2006 - at the beginning of 2007.

Thermal Modeling of the Sandia Flinabe (LiF-BeF$_2$-NaF) Melt Experiment

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An experiment at Sandia National Laboratories confirmed that a ternary salt salt (Flinabe, a ternary mixture of LiF, BeF$_2$ and NaF) had a sufficiently low melting temperature (~305°C) to be useful for first wall and blanket applications using flowing molten salts that were investigated in the Advanced Power Extraction (APEX) Program.[1] In the experiment, the salt pool was contained in a stainless steel crucible under vacuum. One thermocouple was placed in the salt and two others were embedded in the crucible. The results and observations from the experiment are reported in the companion paper.[2] The paper presented here will cover a 3-D finite element thermal analysis of the salt pool and crucible. The analysis was done to evaluate the thermal gradients in the salt pool and crucible and to compare the temperatures of the three thermocouples. One salt mixture appeared to melt and to solidify as a eutectic with a visible plateau in the cooling curve (i. e, time versus temperature for the thermocouple in the salt pool). This behavior was reproduced with the thermal model. Cases were run with several values of the thermal conductivity and latent heat of fusion to see the parametric effects of these changes on the respective cooling curves. The crucible was heated by an electrical heater in an inverted well at the base of the crucible. It lost heat primarily by radiation from the outer surfaces of the crucible and the top surface of the salt. The primary independent factors in the model were the emissivity of the crucible (and of the salt) and the fraction of the heater power coupled into the crucible. The model was "calibrated" using (thermocouple) data and heating power from runs in which the crucible contained no salt.

[1] R.E. Nygren et al., A Fusion Reactor Design with a Liquid First Wall and Divertor, to be published in a special issue of Fusion Engineering and Design
[2] T. J. Lutz, T. J. Tanaka,,J.M. McDonald R.E. Nygren, K. P. Troncosa, T.J. Boyle, and M.A. Ulrickson, Measurement of the Melting Point Temperature of Several Lithium-sodium-beryllium Flouride Salt (Flinabe) Mixtures, this conference

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[1] Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company for the United States Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000

Power-balance control by Slug Tuner for the 175MHz RFQ linac in IFMIF project

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International Fusion Materials Irradiation Facility (IFMIF) is an accelerator-based neutron irradiation facility employing deuteron-Lithium (D-Li) stripping reaction, to produce neutron field similar to the D-T Fusion reactor (2MW/m^2, 20 dpa/year for Fe). In the IFMIF, 40 MeV deuteron beam with a current of 250 mA is injected into the liquid lithium flow with a velocity of 20 m/s. The required current of 250 mA is realized by two beam lines of 125mA, and the output energies at injector, radio-frequency quadrupole (RFQ) linac and drift tube linac (DTL) are designed to be 0.1,5 and 40 MeV, respectively.

The operation frequency of 175MHz was selected for linacs to accelerate the large current of 125mA. For this low frequency, the transmission line using rectangular waveguide is needs to be as large as 1.0 x 0.5m, and accordingly the RF-Input coupler using Iris type also becomes large to inject RF power into RF cavities. Therefore, an RF-Input coupler using a loop antenna with co-axial transmission line has been developed. A multi-drive configuration is necessary to inject a total RF power of 2.3 MW for the RFQ.

For tuning the RFQ, the RF power-balance control by slug tuner is indispensable because the loop antennas and pick-up coils are to be installed into the RFQ cavity, and the RF power-balance in each cavity for quadrupole operation mode (TE_{210}) will be affected by the insertion of these elements. In this study, RF power-balance effects by slug tuners have been measured by using an aluminum low power test module of 175MHz RFQ mock-up with a total length of 1.1m.

The test module consists of a central piece and two end-plate pieces, and several types of loop antennas with different shapes were tried to obtain the optimal insertion depth, 3cm, for realizing appropriate phase differences of TE_{210} mode. Under this condition, cylindrical slug tuners having the various diameters, 3 to 5 cm, were inserted, and S_{21} parameters in each cavity were measured. As a typical result, RF power-balance control less than 20% error can be achieved in case of ϕ3 cm tuner up to 3cm insertion. The obtained results will be a good technical base for RF power-balance control in the IFMIF 175MHz RFQ.

Measurement Of The Melting Point Temperature Of Several Lithium-Sodium-Beryllium Fluoride Salt (FLiNaBe) Mixtures

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The molten salt Flibe, a combination of lithium and beryllium flourides, was studied for molten salt fission reactors and has been proposed as a breeder and coolant for the fusion applications. $2LiF\text{-}BeF_2$ melts at 460°C. $LiF\text{-}BeF_2$ melts at a lower temperature, 363°C, but is rather viscous and has less lithium breeder. In the Advanced Power Extraction (APEX) Program, concepts with a free flowing ternary molten salt for the first wall surface and blanket were investigated.[1] The molten salt (FLiNaBe, a ternary mixture of LiF, BeF2 and NaF) salt was selected because a melting temperature below 350°C that would provide an attractive operating temperature window for a reactor application appeared possible. This information came from a Russian binary phase diagram [2] and a US ternary phase diagram [3] in the 1960's that were not wholly consistent. To confirm that a ternary salt with a low melting temperature existed, several combinations of the fluoride salts, LiF, NaF and, BeF_2, were melted in a small stainless steel crucible under vacuum. The proportions of the three salts were selected to yield conglomerate salts with as low a melting temperature as possible. The temperature of the salts and the crucible were recorded during the melting and subsequent re-solidification using a thermocouple directly in the salt pool and two thermocouples embedded in the crucible. One mixture had an apparent melting temperature of 305°C. Particular attention was paid to the cooling curve of the salt temperature to observe evidence of any mixed intermediate phases between the fully liquid and fully solid states. The clarity, texture, and thickness were observed and noted as well. The test system, preparation of the mixtures, and the melting procedure are described. The temperature curves for the melting and cooling of each of the mixtures are presented along with the apparent melting points. Thermal modeling of the salt pool and crucible was also done and is reported in a separate paper.

[1] R.E. Nygren et al., A Fusion Reactor Design with a Liquid First Wall and Divertor, to be published in a special issue of Fusion Engineering and Design.

[2] N.A. Toropov, I.L. Shchetnikova, Modelnye Sistemy – $Na_2BeF_4\text{-}Li_2BeF_4$ I $Ca_2SiO_4\text{-}Mg_2SiO_4$, Zhurnal Neorganische Khimi 2,8 (1957) 1855-1863; phase diagram.

[3] R.E. Moore, C.J. Barton, W.R. Grimes, R.E. Meadows, L.M. Bratcher, G.D. White, T.N. McVay, L.A. Harris, Phase Diagrams of Nuclear Reactor Materials, R.B. Thoma ed., Oak Ridge National Laboratory, ORNL-2548 (1959) 43; phase diagram also in E.M. Levin, C.R. Robbins, H.F. McMurdie (M.K. Reser, ed.), *Phase Diagrams for Ceramists*, American Ceramic Society, (1979) 436.

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[*] Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company for the United States Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Investigation of Hydrodynamic Instabilities in Shock-Accelerated Flows for ICF

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In inertial confinement fusion (ICF) implosions, the development of a mixing layer at the fuel/ablator interface is driven by shock-induced hydrodynamic instabilities. These instabilities cause perturbations on the interface to grow in amplitude, deform, and eventually develop into turbulent mix. Because such behavior can result in substantial mixing between the compressed hot fuel and outer cold fuel and ablator material, the efficiency and yield of the reaction can be degraded. Thus, it becomes important to understand the physics of these instabilities, and, in particular, the amplitude growth and mixing rates, in order to design the capsules to ensure maximum possible burn efficiency.

Current experiments at the Wisconsin Shock Tube Laboratory (University of Wisconsin-Madison) are investigating these instabilities using various types of perturbed material interfaces in a strongly-shocked gas medium. These studies involve the measurement of instability growth rates, characterization of the unstable flows, and correlation of these characteristics with experimental parameters such as shock strength, material density ratio, and interface geometry and dimensions. These are made possible mainly by planar laser induced fluorescence (PLIF) in one gas species. Experimental studies are coupled with computational modeling that is used to optimize experiment design and to predict and analyze experimental results along with feed back to the computation models.

Recent experiments with a spherical soap bubble have been performed in the vertical, square, large internal cross section (25 x 25 cm) shock tube. The initial interface is three-dimensional, with axial symmetry being a reasonable assumption for a free falling bubble. The strength of the accelerating shock is in the range 2<M<4. The interface is imaged once immediately before shock arrival and twice after interaction with the shock wave, so that the initial conditions are known and the growth rates can be calculated for each experiment.

Concurrently, the *Raptor* code is being used to numerically simulate the shock tube runs for the purpose of optimizing the design of the experiments and for comparing the experimental and numerical results. The code solves the compressible Navier-Stokes equations using a piecewise linear method (PLM) combined with adaptive mesh refinement (AMR).

This paper discusses the progress of the experiments and the computational effort. Examples of the post shock images are shown below.

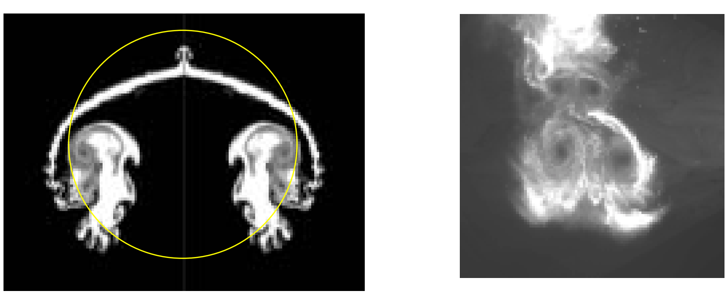

Figure 1. Computational and experimental image of a shocked bubble 0.99ms after impact with a 2.14 Mach shockwave.

Visual Tsunami:
A versatile, user-friendly radiation hydrodynamics design code

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Gas dynamics phenomena in thick-liquid protected inertial fusion target chambers have been explored since the early 1990's with the help of a series of simulation codes known as TSUNAMI. The code has recently been entirely rewritten to make use of modern programming techniques and languages, refine its ability to model thick-liquid protected chambers, expand its capability to a larger variety of systems, and improve its user-friendliness.

The hydrodynamics core model---Euler equations for compressible flows---is retained from previous versions of TSUNAMI. The numerical scheme is upgraded to a faster and more robust scheme. Efficient models for open, reflective and condensing/evaporating boundaries are developed and implemented. For the first time, a two-temperature radiation diffusion model is incorporated into a two-dimensional version of TSUNAMI. Initial condition models are revisited and, in particular, the traditional instantaneous cohesive energy ablation model is upgraded [1]. Each model is benchmarked individually, while the overall set of assumptions is validated with the help of an original gas dynamics experiment discussed in a companion paper [2].

Emphasis has been put on the user-friendliness of the code. The input file builder is now graphics-based, with a highly versatile mesh generator. The geometry, the initial and boundary conditions can be specified through any standard CAD software package. The output processor is an order of magnitude faster than its predecessor. A graphical user interface makes processing output data intuitive and straightforward. The new code is called “Visual Tsunami” to emphasize the programming language of its core, Fortran 95, as well as its graphics-based input file builder and output processor. It is aimed at providing a user-friendly design tool for systems for which transient gas dynamics phenomena play a key role.

[1] C.S. Debonnel et al., “Revisited ablation modeling for IFE chamber design,” these proceedings
[2] C.S. Debonnel et al., “Validation of the Tsunami code through the Condensation Debris Experiment,” these proceedings.

Validation of the Tsunami code through the Condensation Debris Experiment

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Since the early 1990's, TSUNAMI has been the main simulation code used to model gas dynamics in thick-liquid inertial fusion energy (IFE) target chambers. Other applications include modeling the National Ignition Facility chamber and mini-chamber. Typically, gas dynamics systems modeled with TSUNAMI have had three distinct temporal phases: the high energy density plasma physics stage, including laser and/or heavy-ion beam energy deposition; the hydrodynamics expansion phase; and the late time period during which condensation takes place, in-flight and/or onto surfaces. The TSUNAMI code is aimed at modeling the second phase, making use of information from the first, and providing input to model the third. TSUNAMI has always relied on the assumption that high-energy density physics phenomena happen on short time scales, and only a few integrated effects need to be taken into account. Simple scaling laws or coarse mapping of detailed plasma physics code output were said to provide relevant initial conditions. While various models of the code, such as the hydrodynamics core and the ablation models have been benchmarked against experiments or other codes, the overall set of assumptions has never been validated against a relevant experiment. The novel Condensation Debris Experiment (CDE) constitutes the first "integrated" validation of TSUNAMI. CDE presents the aforementioned three phases and other similarities with IFE: a point-like source of hot debris, x-ray induced ablation, expansion of the gas into a shaft and the chamber, and then, ultimately, its condensation in-flight and on the walls of the chamber. CCD pictures and pressure measurements from the first two CDE experimental campaigns, performed on the Z laser facility at Sandia National Laboratory in New Mexico, are used to validate "Visual Tsunami," the latest version of the code [1]. Agreement between the code and the experimental results is satisfactory and confirms the ability of Visual Tsunami to be used as a design tool for gas dynamics systems.

[1] C.S. Debonnel et al., "Visual Tsunami: A versatile, user-friendly radiation hydrodynamics design code," these proceedings

Practical ablation models for IFE chamber design

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Inertial Fusion Energy (IFE) targets will emit a considerable amount of energy in the form of x-rays. Those x-rays will deposit their energies in small layers of the target-facing material, vaporizing some of it. This is referred to as "ablation." In thick-liquid protected chambers, ablated mass and energy are essential in assessing the total impulse imparted to the liquid structures through the rocket-effect impulse, gas venting time, and pocket pressurization.

The Berkeley radiation hydrodynamics code "Visual Tsunami" [1] makes use of the open literature Lawrence Livermore EPDL---the Evaluated Photon Data Library [2]. Various ablation models are reviewed and implemented; also, differences are illustrated using a typical thick-liquid molten salt pocket. A new instantaneous model is proposed for ablation of high heat capacity molten salts. This new model justifies the traditional energy redistribution model, which assumes that all the x-ray energy deposited in the region between the cohesive and saturation depths is used to vaporize the molten salt. An extensive comparison of instantaneous ablation models to experimental results and other ablation codes is performed.

[1] C.S. Debonnel et al., "Visual Tsunami: A versatile, user-friendly radiation hydrodynamics design code," these proceedings
[2] D.E. Cullen et al., "The 1989 Livermore Evaluated Photon Library (EPDL)," UCRL-ID-103424, Lawrence Livermore National Laboratory.

Target Injection Tracking and Position Prediction Update*

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To achieve high gain in an inertial fusion energy power plant, driver beams must hit direct drive targets with ±20 μm accuracy (±100 μm for indirect drive). Targets will have to be tracked with even greater accuracy. The conceptual design for our tracking system, which predicts target arrival position and timing based on position measurements outside of the reaction chamber was previously described [1]. The system has been built and has begun tracking targets at the first detector station. Additional detector stations are being modified for increased field of view. After three tracking stations are operational, position predictions at the final station will be compared to position measurements at that station as a measure of target position prediction accuracy.

The as-installed design will be described together with initial target tracking and position prediction accuracy results. Design modifications that allow for improved accuracy and/or in-chamber target tracking will also be presented.

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Fabrication of a Full Density Polyvinylphenol Overcoat for IFE Targets*

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We are performing research and development to evaluate methods of overcoating divinylbenzene (DVB) foam shells with a full density polyvinylphenol (PVP) overcoat approximately 3–5 μm thick for use as inertial fusion energy (IFE) targets. We have focused our initial efforts on optimizing the chemistry of the PVP polycondensation reaction to better link the full density overcoat with the DVB foam shell, as well as improving the smoothness of the overcoat surface. Initial study is focused on the fluid exchange process utilized in the polycondensation reaction to create the PVP overcoat and the supercritical CO_2 drying process.

Additionally, we have applied chemical engineering principals and techniques to begin to increase the scale and improve the yield of the overcoat process. These techniques include the use of optical imaging devices and flow control monitors to control rate and dimensions of shell formation, as well as advancing the current laboratory bench scale process used for fluid exchange for both the DVB foam shell and PVP overcoat into a continuous process via a flow through tube design, under development in collaboration with Schafer Corporation at Sandia National Laboratory. This research is intended to develop the basis for high yield, high reproducibility IFE target mass production.

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Rep-Rated Target Injection for Inertial Fusion Energy*

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Inertial Fusion Energy (IFE) with laser drivers is a pulsed power generation system that relies on repetitive, high-speed injection of targets into a fusion reactor. To produce an economically viable IFE power plant the targets must be injected into the reactor at a rate of 5 to 10 Hz.

To survive the injection process, direct drive (laser fusion) targets (spherical capsules) are placed into protective sabots. The sabots separate from the target and are stripped off before entering the reactor chamber. Indirect drive (heavy ion fusion) utilizes a hohlraum surrounding the spherical capsule.

In our target injection demonstration system, the sabots or hohlraums are injected into a vacuum system with a light gas gun using helium as a propellant. To achieve pulsed operation a rep-rated injection system has been developed. The system will allow bursts of up to 12 targets at 6 Hz. Using the current system single shot tests have been successfully run with direct drive targets to show sabot separation under vacuum and at barrel exit velocities of ~350 m/s. These tests have also included three shot bursts of direct drive targets at a rate of approximately 5 Hz.

The next step in demonstrating the rep-rated injection system will be to perform a full 12 shot burst at 6 Hz. To achieve this goal design modifications to provide tighter tolerances and alignment in the revolver-based feed system have been made. An additional modification for this system, to be implemented at a later date, will be to incorporate simultaneous revolver loading and shooting to move beyond burst operations and show viability of long-term rep-rated operation.

The details of the existing revolver feed system along with operational data will be presented.

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Target Physics Scaling for Z-Pinch Inertial Fusion Energy

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Recent improvements in the technology of fast pulsed electrical power and load design have led to a growing interest in the use of fast z-pinches as x-ray sources for inertial confinement fusion (ICF) energy applications[1]. Fast z-pinch implosions of a high-Z plasma can efficiently convert the stored electrical energy in a pulsed power accelerator into high temperature dense plasmas that radiate as soft x-ray sources. In *dynamic hohlraum* x-ray sources, cylindrical arrays of tungsten wires are used to form a z-pinch imploding plasma shell[2], which generates and traps x-rays as it as it stagnates on a low opacity cylindrical target centered on the z-pinch axis. The trapped radiation can be either directed out of the *dynamic hohlraum* and into a *static-walled hohlraum* (as has been demonstrated in the experiments described in Ref. 3), or can be used to directly implode an ICF capsule located within the *dynamic hohlraum* (as has been demonstrated in the experiments described in Ref. 4). In recent experiments, ~ 10^{10}-10^{11} D-D neutrons have been produced via the implosion of a capsule located directly within a *dynamic hohlraum*[4]. In the present paper, we will provide descriptions of the ongoing ICF experiments, concepts for future ignition and high yield target designs, and an evaluation of the ICF target physics and z-pinch scalings required for the eventual development of a high gain Z-pinch IFE target.

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Large-Area Electron Beam Diode and Gas Cell Design for a KrF Laser IFE System*

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In a KrF laser inertial fusion energy (IFE) power plant [1] an array of *N* laser amplifiers would focus and symmetrically illuminate a fusion target with a total of ~3 MJ of laser light. The lasers are pumped by large-area electron beam diodes, whose technology is presently being developed on the Electra [1] and Nike [2] facilities at the Naval Research Laboratory in Washington, DC. Sufficient progress has been made on these facilities that we can start evaluating the laser amplifier concepts for a fusion power plant. We are currently evaluating two such concepts where both have an the amplifier that is pumped by two opposing arrays of electron beams that are driven by solid state pulsed power systems. One concept has a 30 kJ laser output and is pumped with two large electron beams. The other is a 50 kJ design that is pumped by two arrays of four electron beams. Each beam is injected through its own foil support structure (hibachi) into a common laser cell. The former “monolithic cathode” (30 kJ case) minimizes KrF physics risks and has potentially higher efficiency; whereas the latter “segmented approach” (50 kJ case) minimizes the pulsed power costs and stresses on the foil support (hibachi).

We have carried out a preliminary design study of the electron beam transport for both these concepts using large-scale numerical simulations with the particle-in-cell code LSP [3]. The simulations include detailed geometric representations of the cathode, foils, and hibachis as well as gas transport models. This integrated modeling procedure has been successfully benchmarked against recent Electra large-area diode experiments [4,5]. The simulations provide estimates of the magnitude and spatial distribution of the electron beam energy deposited in the gas, foils, and hibachi. The estimates of energy deposition in the laser gas from the simulations will be integrated with existing system designs.

The amplifier performance has been examined with Orestes [6], a first principles KrF physics code that self-consistently couples the various physical processes such as e-beam pumping and amplified spontaneous emission. Orestes has been successfully benchmarked against Electra oscillator experiments [7] for a number of KrF gas compositions and pressures.

* Supported by the U.S. DOE/NNSA.

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Electron Injection for Space-Charge Control in HIBF

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One key issue in heavy ion beam fusion (HIBF) is how to effectively focus high-current and high-perveance ion beams onto a small target area, usually with a radius of 2-5mm. Space charge neutralization must be achieved to prevent defocusing as the ion density compresses during focusing. The present study is concerned with the physics of space-charge neutralization by use of various concepts for injection of electrons into the ion beam in addition to plasma neutralization. To date the most widely studied approaches to neutralization use a preformed plasma or ionize a background gas in the beam line. However, these schemes alone face difficulties due to local non-uniformities in charge neutralization due to beam dynamic/focusing effects. Thus, two supplemental techniques are under study and will be discussed here. One involves axially injection of electrons created in a thin foil in the beam path. The other uses a magnetic field to guide the electrons into the direction of the beam path. Both techniques would be used in combination with conventional background plasma, providing added control. A preliminary analysis of these schemes will be presented. Both analytical method and computational simulation using particle-in-cell technique are applied to solve simplified three-fluid equations in this paper. The required electron profile and focusability of the ion beam resulting from the calculation will be given. In addition, the design of possible components that could be added onto an existing HIBF experiment such as NTX at LBL to provide a proof-of-principle test of this type of electron injection will be discussed.

Experimental study of voltage breakdown over flibe liquid surfaces for IFE applications

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The beryllium fluoride and lithium fluoride eutectic mixture denominated flibe has been proposed as main option for coolant, breeder and chamber protection material in several IFE power plant design and concept studies. They involve situations in which the molten salt liquid surface is exposed to a high, transient electric field generated by space charges (as for the Heavy-Ion Beam lines of HYLIFE-II) or currents (as in the Z-pinch power plant concept). Voltage breakdown is generally inhibited by a vacuum gap allowed by the low vapor pressure of flibe in equilibrium over the liquid surface.

This paper presents the results of the experimental work on voltage breakdown over a flibe liquid surface performed at UCLA. A stagnant pool of flibe is maintained in thermal equilibrium in a vacuum chamber. The pool is contained inside an electrically grounded nickel crucible that functions as the discharge anode when breakdown occurs. A transient electric field is generated by charging an electrode suspended over the pool surface up to 12 kV in less than 0.6 μs. Breakdown voltage is measured as a function of the electrode distance from the surface and the liquid temperature. The composition of the gas in equilibrium with the liquid surface is monitored with a residual gas analyzer. The effect of the impurities dissolved in the molten salt on voltage breakdown is discussed along with the relevance of the presented results for IFE applications.

Characterization of Arc Generated Plasma Interactions with a Liquid Metal Medium

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The study of the interaction of plasma with liquid metals is a key component to future fusion devices design. With liquid metals being considered as inner wall material for both torodial and inertial confinement fusion devices it is important to be able to predict the way in which the liquid metal wall will interact with the fusion plasma. This study focuses on the plume of liquid metal created when the plasma impacts on the liquid metal medium and the results will be applicable in inertial confinement design. The liquid metal plume will be characterized using data on the particulate liquid metal colleted on aluminum buttons positioned in various locations; with this information plume shape, density and velocity can be determined. Also of interest is the size of the particulate matter ejected from the liquid metal medium.

Data on the plume is collected using the “Arc Generated Explosion Impact on Substrate” (AGEIS) device. This device is a metal cylinder with multiple ports for the two separate electrodes, fiber optic line, pressure regulator and gage, and an electrical port for the heating element wires and the thermocouple wires. The internal structure of the cylinder consists of a metal box frame that holds the heating plate element at the bottom of the cylinder and has adjustable arms attached with collector buttons. The plasma is generated using a taut copper wire stretched between the two electrodes; a large capacitor bank is used to pulse energy into the wire vaporizing it and creating a copper plasma which is further sustained by arcing between the two electrodes. A liquid lead pool is positioned on the heating plate such that the pool surface is 1.5cm below the copper wire. The wire length is 5cm long making it wider than the pool diameter so that the electrode tips don’t interfere with the plume formation. A fiber optic line is positioned to collect spectrum data on both the wire and the pool. The current and voltage input into the wire are recorded in time using a high voltage probe and a Pearson coil for the current. The electrical data along with the spectrum data allow for the characterization of the plasma that impacts on the liquid lead pool. The entire cylinder is maintained at a pressure of 20 Torr because lowering the pressure causes the breakdown of the air in the chamber when the power is pulsed into the system. For the characterization of the plume, multiple collector buttons are used in each “shot” with the same parameters for the system and then moved to cover different geometries and velocities. A scanning electron microscope is used to identify the size and density of the particulate matter on the buttons and using multiple locations for the buttons with the same system parameters will allow for the plume to be fully modeled. System parameters can then be changed and the process repeated to determine which system parameter changes affect the plume formation.

Liquid Vortex Shielding for Fusion Energy Applications

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Annular vortex flows can protect critical components in heavy-ion fusion (HIF) chambers, as well as fusion chambers for other inertial and magnetic fusion chamber applications. Such liquid vortexes have been developed for protecting heavy-ion beam tubes in the HYLIFE chamber, by forming a layer of liquid that coats the inside of the tube near the beam port. This annular liquid jet, formed in the vortex tube, provides additional neutron shielding for the final focus magnet and condenses vaporized molten salt that may enter the beam tubes. To maximize their effectiveness the vortex tube flow must have a smooth surface while being turbulent to promote high condensation rates. Particle Image Velocimetry (PIV) measurements that permit surface topology and turbulence structure characterization are being conducted to study the characteristics of turbulence in these vortex tubes.

Experiments to understand larger scale vortex flows, which may eventually serve as blankets to protect first walls in advanced inertial and magnetic fusion chambers, are also being conducted.

For optical measurement like PIV it is important to match the index of refraction of the fluid with the tube material. Here the novel use a low viscosity mineral oil, with an index of refraction matched to acrylic, is reported. The oil thermophysical properties are ideal for simulating molten-salt fluid mechanics, since they permit the Reynolds, Froude and Weber numbers to be matched simultaneously in experiments of around 25% of full scale size.

Thermal Loading of a Direct-Drive Target in Rarefied Gas

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In an inertial fusion energy (IFE) power plant, each fusion micro-explosion (~10 Hz) causes radiation and pressure shock waves that impose extreme loads on the IFE reactor wall and driver optics. This deposition of energy in the reactor wall over a short time could result in the sputtering of wall material, high stresses, and fatigue. The loading on the wall must remain sufficiently low to ensure that economic and safety constraints are met.

One proposed method for decreasing the intensity of the wall loading is to fill the reactor chamber with a gas, such as Xe, at low density. The gas will absorb much of the radiation and ion energy from the fusion event, and then slowly release it to the chamber wall. Unfortunately the protective gas introduces major heat loads on the direct-drive target. The thermal loading of a target, during injection, largely determines the viability of that target upon reaching chamber center (CC).

The objective of this work is to quantify and characterize the heat flux resulting from the interaction of the target and the protective gas. For the range of expected protective gas densities (3e19 m^{-3} to 3e21 m^{-3}), the energy exchange takes place in the molecular or transition regime. The high Knudsen number flow (Kn ~ 1-100) around the target is modeled using DS2V (a DSMC program). Using DS2V, this work explores the affect of the protective gas density, temperature, sticking (condensation) and accommodation coefficients on the heat flux to the target.

As expected the heat flux is extremely sensitive to the density of the protective gas (increasing as the gas density increases). It is also found that the sticking and accommodation coefficients significantly influence the heat load on the target. Unfortunately the sticking and accommodation coefficients are unavailable for the temperatures expected in the reaction chamber; thus, an experimental determination of these coefficients is necessary.

Modeling DT Vaporization and Melting in a Direct-Drive Target

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During injection, inertial fusion energy (IFE) direct-drive targets are subjected to heating from energy exchange with the background gas and radiation from the wall. This thermal loading could cause phase change (vaporization and/or melting) of the deuterium-tritium (DT). In the past, it was assumed that any phase change would result in a violation of the stringent smoothness and symmetry requirements imposed on the target. The objectives of this work are to demonstrate the advantages, and determine the effect on target symmetry of allowing phase change under different assumed scenarios.

This work summarizes the results from a one-dimensional finite difference model that was created to simulate the coupled thermal and mechanical response of a direct-drive target to an imposed heat flux. The model utilizes a simple method, known as the apparent c_p method, to account for the effect of melting on the heat conduction in the target. The model also accounts for the change in density, and hence deflection of the polymer shell and DT solid, attendant with DT vaporization and melting.

It is found that as the amount of melting and vaporization increases, the ultimate strength of the DT solid and/or polymer shell can be exceeded. The amount of liquid and vapor is given for several heat fluxes as a function of time. Insight is gained as to the fertility, in relation to vapor bubble nucleation and growth, of the target environment by calculating the maximum superheat in the liquid DT. It is also shown that under certain circumstances preexisting vapor can be minimized or eliminated by allowing melting. The results of this preliminary study suggest that some phase change may be allowable. To gain a better understanding of the consequences of phase change it is proposed that a new 2-dimensional model be created and coupled with experiments.

Neutronics Investigation into Lithium/Vanadium Test Blanket Modules

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Technical discussion on ITER-Test Blanket Modules (TBM) has been made in Test Blanket Working Group (TBWG). The discussion is based on the concepts of breeding blankets to be developed for DEMO. The present concepts of lithium self-cooled blanket are categorized into "No beryllium (Li/V)" and "With beryllium (Li/Be/V)" concepts. The former and the latter systems use neutron reaction with ^{7}Li and Be for enhancing the Tritium Breeding Ratio, respectively. It has been shown that, with appropriate designs, tritium breeding self-sufficiency will be satisfied for either systems. The DEMO blanket and TBM for Li/Be/V concepts have been investigated by Russian R&D program and Detailed Description Document for the TBM is available.

The Li/V concept has some advantages over Li/Be/V concept; (1) the blanket structure can be simplified, (2) the system is free from the issues of natural resource limit and handling safety concerning beryllium, (3) no periodic replacement of blanket material is necessary. The present study is the initial effort of investigating Li/V TBM from the neutronics aspects.

The primary purpose of the module test was defined as validation of the tritium production rate prediction carried out based on the neutron transport calculation. For this purpose the module was designed to be composed of sectioned thick boxes which accommodate slow tritium flow. This system enables to measure the tritium production rate as a function of the distance from the first wall. The size of the boxes was limited so as to satisfy the introduction limit of liquid lithium into the test port.

In this study, the neutron energy spectrum and tritium production rate have been calculated using MCNP code with JENDL-3.2 database. The available ITER-FEAT structure was used as the system for the calculation of the module. In addition, ITER-FEAT structure with all the plasma-facing area replaced with Li/V breeding blanket (a hypothetical Li/V reactor) was used for comparative calculation.

A comparison was made of the neutron spectrum in ITER-TBM and the Li/V reactor, and showed that the flux of low energy neutrons, including thermal neutrons, is significantly higher in ITER-TBM than that in the Li/V reactor. Thus it is possible that the tritium production by reaction of ^{6}Li with thermal neutrons should be enhanced in ITER-TBM. As a result the contribution of ^{7}Li to the overall tritium production in the module will be significantly lower than that in the reactor blanket. For better simulation for the reactor environment, it is necessary to shield thermal neutrons in the TBM area. For this purpose, a calculation was made for the case in which the module was covered with a B_4C layer. It was shown by the calculation that the spectral tailoring is possible by adjusting the thickness of the B_4C layer. Predicted tritium production rate was obtained in the spectrum-tailored module.

Coating with thin tungsten layer was considered as protection of the first wall. The effects of the coating to the reactor TBR and the tritium production rate of the module were estimated.

Estimation of Radioactivities in the IFMIF Liquid Lithium Loop due to the Erosion and Corrosion of Target Back-wall

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International Fusion Materials Irradiation Facility (IFMIF) is an intense neutron source for testing materials that will be used to construct fusion reactors. Neutrons are generated through deuteron-lithium (Li) stripping reactions. In the design of the facility, the liquid Li target will be bombarded by deuterium beam with the power at 10 MW to produce a great amount of neutrons so that materials will be irradiated up to the level of 50 dpa/y. Through this process, a mass of radioactive remnant ^{7}Be is produced in the liquid. In addition, surrounding structural materials are extremely activated by the neutrons, which causes several adverse effects for the maintenance work. One of the effects is caused by erosion and corrosion of the target back-wall, through which activated compositions move into liquid lithium. Since the lithium flows in pipes along the lithium lines, the activated erosion/corrosion products are transported everywhere in the loop and accumulate as the operation goes on. The radiation dose rate becomes higher as the activation build-up grows, and the level drastically affects the strategy for the maintenance works around the Li loop. The half- lives of the activities are relatively large compared with reaction remnants, and the effect has not been studied yet in the IFMIF design. In the present work, the radioactivity in the Li loop was estimated and the influence on the maintenance scenario was assessed.

The neutron spectrum in the back-wall was calculated by McDeLicious, an extended version of a Monte Carlo transport code MCNP, to sample source neutrons based on the deuteron-lithium reaction cross section data. The activation reaction rate of the back-wall was calculated based on the IEAF-2001 library, the latest version of nuclear activation data in the intermediate energy range up to 150 MeV compiled in Europe for the design of accelerators. The activation build-up during the operation and the decay after shutdown were calculated by the ACT-4 code of the THIDA-2 system, a design code system for fusion reactors developed in JAERI. The material of the target back-wall is the stainless steel type 316, and the total amount of radioactivity in the back-wall was calculated to be 2×10^{18} Bq/m^3 (several tens Ci/cc) one month after the shutdown. The radioactivity in the IFMIF Li loop was evaluated under the erosion and corrosion rate at 1 μm/y based on the data in FMIT project. The area of stainless steel that suffers erosion and corrosion is 100 cm^2, which is as large as the beam footprint on the back-wall. Thus, the total amount of the activity after one-year operation was calculated to be 2×10^{10} Bq (some 1 Ci). On the other hand, the amount of the deuteron-lithium reaction remnant ^{7}Be would be as much as 4.5×10^{15} Bq (about 10^5 Ci) during the maintenance work according to the result of the IFMIF design activity. As a result, the radioactive compositions in Li loop supplied by the erosion and corrosion of the target back-wall turned out to be negligibly small compared with that of reaction remnants.

Neutronics experiments using small partial mockup of the ITER test blanket module with solid breeder

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In fusion DEMO reactors, the blanket is required to provide a tritium breeding ratio (TBR) of more than unity by neutron induced reactions in lithium. In order to verify the accuracy of the tritium production rate (TPR) and also to study the impact of the tungsten armor on the TPR experimentally, neutron irradiation experiments have been performed with a small partial mockup relevant to the ITER test blanket module proposed by JAERI using DT neutrons at FNS of JAERI.

A solid breeder blanket mockup, composed of a set of slabs of 16 mm thick F82H, 12 mm thick Li_2TiO_3 (^6Li enrichment of 40 %) and 200 mm thick Be with about 660 mm height and about 660 mm width each, was installed, and DT neutron irradiation experiments were conducted. In the experiments three types of mockups were tested: without the armor; with 12.6 mm thick W armor; and with 25.2 mm thick W armor. Two campaigns were also performed for the mockup without the armor. A reflective enclosure made of SS was installed surrounding the DT neutron source in one campaign. It can take into account the effect of the incident back-scattering neutron source. It was not installed in the other campaign. Fifteen slices of Li_2CO_3 pellets, with 13 mm in diameter and 0.5, 1, and 2 mm in thickness, were embedded inside the Li_2TiO_3 slab. After the irradiation, induced radioactivities were measured by beta ray intensity of these pellets with a liquid scintillation counter, and the TPR was evaluated.

Numerical analyses were conducted by using MCNP-4C with FENDL-2. The calculation results agree well with the experiment ones within 13 and 2 % for the campaign with and without the enclosure, respectively. From this study, it can be clarified that the TPR can be predicted in very high accuracy for the case without the enclosure. Uncertainties with the enclosure are larger than those without one. It is expected that this occurs, due to the uncertainties of the cross section data about the back-scattering neutron. There are a number of back-scattering neutrons entering the blanket in the actual fusion reactor. The contribution of the back-scattering neutrons in the present experiment with the enclosure is larger than that in the actual fusion reactor, so it can be concluded that the prediction accuracy is in the range of 2 – 13 % in the calculation for the actual fusion reactor.

From the experiment with the 12.6 mm and 25.2 mm thick tungsten armors, the integrated TPRs were reduced by 8 and 3 % relative to the case without the armor, respectively. The fast and thermal neutron fluxes are reduced by the tungsten, so it can be concluded that the TPRs were reduced by installing the tungsten armors. In the blanket design proposed by JAERI, it is expected that the reduction of the TBR is less than 2 % as the thickness of the W armor is less than 5 mm.

Tritium Self-Sufficiency and Neutron Shielding Performance of Liquid Li Self-Cooled Helical Reactor

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The concept of a liquid Li self-cooled reactor has various attractive aspects such as high thermal conductivity, simple blanket structure, no irradiation damage to the breeder material etc. Moreover, one of the unique features is possibility of tritium self-sufficiency without beryllium neutron multiplier. The Force Free Helical Reactor conceptual design (FFHR-2) has been carried out with a self-cooled Flibe blanket system [1]. One of the issues of the Flibe concept is the compatibility between the tritium self-sufficiency and the performance of neutron shielding for super-conducting magnets within the limited blanket dimension. The purpose of the present study is to investigate the tritium-sufficiency and the shielding performance of the FFHR-2 with a liquid Li self-cooled blanket system.

The FFHR-2 has the major radius of 10.0 m and the average plasma radius of 1.2 m. Neutron wall load of 1.5 MW/m^2 is considered in the design. For the Flibe blanket with the total thickness of 90 cm, the local TBR of ~1.4 is expected. The fast neutron flux at outside of the shield is estimated to be 6.3×10^{10} $n/cm^2/s$.

Neutron transport calculation for the FFHR-2 with a liquid Li self-cooled blanket system was performed using MCNP-4C code and JENDL-3.2 nuclear data library. V-4Cr-4Ti alloy was employed as a structural material for liquid Li breeding channels. Main component for a reflector and a shield was low activation ferritic steel, JLF-1.

In the concept without beryllium multiplier, the blanket region of ~50 cm from the first wall was filled with liquid Li channels. The tritium production was enhanced by enrichment of ^6Li to 25 % for reducing the thickness of the liquid Li layers and improving the shielding performance. A 15cm layer of JLF-1 and a 38cm layer of JLF-1 (70%) + B_4C (30%) were placed as a reflector and a shield. The total thickness of the blanket was assumed to be ~105 cm, which is within the allowable dimension of the FFHR-2. The estimated local TBR and fast neutron flux at outside of the shield were ~1.4 and 5.6×10^{10} $n/cm^2/s$.

The thickness of the liquid Li breeding layers could be drastically reduced by introducing Be multiplier. While the total thickness of the liquid Li channels was decreased to 23 cm, the local TBR of ~1.4 was retained by installing a Be layer of 5 cm and enrichment of ^6Li to 40 %. The fast neutron flux at outside of the shield was estimated to be 9.7×10^9 $n/cm^2/s$, i.e. ~1/5 of the Li/V blanket system without Be multiplier, with a 20 cm layer of JLF-1 and a 57 cm layer of JLF-1 (70%) + B_4C (30%).

From the results of the calculation, it is confirmed that the compatibility between the tritium self-sufficiency and the neutron shielding performance within the limited blanket dimension for the Li/V blanket system could be comparable to that of the Flibe blanket system. Further improvement in the shielding performance is likely to be possible for the Li/V blanket system by introducing Be neutron multiplier.

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Views on Neutronics and Activation Issues Facing Liquid-Protected IFE Chambers

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Over the past few years, the ARIES team has been defining the design space and operational windows for both laser and heavy ion driven inertial fusion energy (IFE) concepts from the viewpoint of a viable power plant system, rather than developing a point design [1]. For the heavy ion beam study, we were concerned with the feasibility of protecting the steel-based structure (advanced ferritic steel or 304-SS) of the chamber with a thick wall of Flibe or Flinabe liquid breeders. Specifically, the concern is the ability of the thick liquid wall to protect the structure during the entire plant life (40 FPY) while providing an adequate tritium supply for machine operation and satisfying the ARIES top-level requirement of generating only low-level waste (Class A or C). In addition, the helium production level is a concern if the design mandates cutting and rewelding the structure and flow nozzles for component replacement or maintenance during operation.

It is found that Flibe is a better breeder than Flinabe, thus a wall thickness (58% breeder and 42% void) of 85 cm is required for Flibe and 150 cm for Flinabe for a breeding ratio of 1.08. While the 200 dpa level at the structure indicates both breeders could protect the advanced ferritic steel (FS) for 40 FPY, the FS reweldability limit of 1 He appm is greatly exceeded, meaning the FS structure and flow nozzles cannot be rewelded at any time during operation.

The activation of the chamber structure is severe as all steel-based alloys generate high-level wastes. A Class C low-level waste cannot be achieved unless the impurities of the FS and 304-SS alloys are strictly controlled. The main contributors to the waste disposal rating are ^{94}Nb (from Nb), ^{99}Tc (from Mo), and, to a lesser extent, ^{192n}Ir (from W). Thus, consideration of advanced FS or 304-SS for ARIES-IFE rests heavily on the assumption that Nb and Mo impurities can drastically be controlled for the “off-the-shelf” materials. The high cost of impurity control must be factored in the unit costs of the modified FS and 304-SS structures. Practically and economically, complete removal of Nb and Mo impurities can never be accomplished. An alternate approach is to thicken the blanket and in the meantime adjust the breeding level by depleting the Li of the breeder. A combination of impurity control and a thicker blanket with depleted Li represents a viable solution to the high-level waste problem of the ARIES-IFE liquid-protected chamber.

Another concern for the liquid-protected chamber is related to the IFE pulsed nature. The high instantaneous damage rate can lead to significant changes in the microstructure of the material. Furthermore, the instantaneous deposition of the neutron energy can cause isochoric heating problems with significant pressure waves that could impact the fatigue life of the structure. Depending on the rep rate, the structure temperature fluctuates 4-6 times per second with internal pressure reaching 100 atm. Pulsing produces significant strain in the structure. Fatigue from repetitive shock waves could shorten structure lifetime and cause internal cracks. The validity of the structure over millions of shots is still an open question.

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Activation Analysis for Two Molten Salt Dual-Coolant Blanket Concepts for the US Demo Reactor

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The US has considered, among other options, two blanket concepts for Demo reactor in which helium is primarily used to cool the first wall (FW) and structure whereas molten salt (MS) is used as both coolant and breeder. The Demo reactor has a maximum neutron wall load, NWL, of 3 MW/m^2 at the Outboard (OB) (~2 MW/m^2 at the IB) and an average NWL of ~2.5 MW/m^2 at the OB (~1.6 MW/m2 at the IB). Conventional reduced activation ferritic steel (RAFS, F82H) is used as the structural material in both blanket concepts. The "Flibe" MS, with mole ratio 1:1 of LiF and BeF_2, is used in the first option with 50% Li-6 enrichment while the "Flinabe", with mole ratio of 1:1:1 of LiF, NaF, and BeF_2, is used in the second option with 60% Li-6 enrichment. The Flibe with the 1:1 mole ratio was chosen due to its lower melting point of ~380^{o}C as opposed to ~470 ^{o}C with a mole ratio of 2:1. Low melting point has the advantage of widening the coolant operating temperature window and hence improving the thermal efficiency of the plant. Flinabe has also lower melting point (~305^{o}C) but its tritium breeding capability, characterized by the attainable tritium breeding ratio TBR, is lower than Flibe. Beryllium is used as a neutron multiplier in both concepts (with larger amount in the case of Flinabe) to improve the local TBR. The blanket in both concepts has a thickness of 65 cm in the OB (40 cm in IB), including a 5 cm-thick front Be zone (8 cm in case of Flinabe). The radial build in both blanket concepts was arrived at after several iterations to maximize the TBR while ensuring adequate radiation protection to the magnet and making the vacuum vessel a life-time component. The FW is estimated to last ~5 years before replacing the blanket. The results of this neutronics assessment are reported in a companion paper.

In This paper we present the results for assessing the radioactivity and decay heat for up to 1000 years after shutdown. This assessment is performed separately for the structural material, the Be multiplier and the breeder (Flibe/Flinabe). The average wall loading on the OB and IB is considered in the analysis and the modules in the poloidal direction are assumed to be ~8.3 m long. In estimating the pertaining parameters in the Flibe (or Flinabe) it is assumed that 1/4 of its volume is inside the reactor during operation with the rest flowing outside. The total activity and decay heat in the F82H structure is very similar in both concepts (~2000 MCi and ~10 MW, at shutdown, respectively). The total activity in the Flibe is slightly larger in the Flibe but the decay heat (dominated by Na-22 and Na-24) is much larger in the Flinabe by up to an order of magnitude in the time frame of 1 hour-10 years after shutdown. The dominant contributor to Flibe decay heat is F-18 up to ~1 day after shutdown. The Class C waste disposal rating (WDR) was estimated for each material using both Fetter and NRC limits. For Flibe, Flinabe and Be the WDR is much lower than unity. However, the WDR for F82H is ~0.6-1.3 (Fetter) and is dominated by Tc-99 and Nb-94. They are attributed to reactions with Mo and Nb present in F82H with levels of 70 wppm and 4 wppm, respectively. To ensure that F82H qualifies for shallow land burial, it is suggested to reduce these two impurities to ~50 and ~3 wppm, respectively. The results cited in this paper are needed to assess safety concerns such as thermal response during accident conditions and the mobilization of the radiological inventories and site boundary dose following these accidents.

On the Strategy and Requirements for Neutronics Testing in ITER

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Neutronics testing is among the several types of fusion technology testing scheduled to be performed in a test facility such as ITER. The three ports assigned for testing in ITER will test several blanket concepts proposed by the various parties with test blanket modules (TBM) that utilize different structural materials, breeders, and coolants. Nevertheless, neutronics issues to be resolved in ITER-TBM are generic in nature and are important to each TBM type. Dedicated neutronics tests specifically address the accuracy involved in predicting key neutronics parameters such as tritium production rate, TPR, volumetric heating rate, induced activation and decay heat, and radiation damage to the reactor components. In addition, neutronics analyses are required to provide input support for other tests (e.g. heating rates for thermo-mechanics tests).

In this paper, we address some strategies for performing the neutronics tests. Tritium self-sufficiency for a given blanket concept can only be demonstrated in a full sector, as envisioned in a Demo reactor, including a closed tritium fuel cycle with its components such as tritium separation and clean-up systems. However, testing in ITER TBM can provide valuable information regarding the main parameters needed to assess the feasibility of achieving tritium self-sufficiency. Unlike the case of using engineering scaling to reproduce demo-relevant parameters in an “Act-alike” test module, dedicated neutronics tests require a “Look-alike” test module for a given blanket concept. This is due to the desire to quantify realistic error bars associated with various neutronics parameters when predictions with various codes/nuclear data are compared to measured data. Furthermore, it is strongly desired to perform neutronics tests on as “cold” module as possible to allow accurate measurement of tritium production. It is therefore recommended to perform these tests as early as possible during the DD operation phase (year 4) or the low duty cycle DT operation phase (years 5 and 6).

The paper also addresses the operational requirement (i.e. fluence requirement) as well as the geometrical requirement of the test module (i.e. minimum size) in order to have meaningful and useful tests. For example, TPR, neutron and gamma spectrum measurements, etc., require fluence in the range of 1-10^6 Ws/m^2. Any linear combination of neutron wall loading and operating time in this range is acceptable, e.g., 400 s (ITER pulse) at 2.5 x 10^{11} n/cm^2 is adequate. Measured neutronics data requires high spatial resolution. This requires the measured quantity to be as flat as possible in the innermost locations inside the test module both in the poloidal and toroidal direction. This sets, in turn, certain geometrical requirement on the size of the TBM. In this paper, we discuss the minimum test module size that makes the predicted data, whose measurements are intended in ITER (e.g. TPR), not to deviate by more than 5% (for improved simulation) relative to the full coverage case found in a Demo or a power reactor. Preliminary results show that a test module of ~45 cm in the poloidal direction and ~60 cm in the toroidal direction may achieve this accuracy goal. The impact of the frame thickness of the test port (cooled with water) on these measurements is also investigated and the results will be reported in this paper.

Three-Dimensional Modeling of Complex Fusion Devices Using CAD-MCNP Interface

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For a commercial power plant fusion device, many engineering design concepts were evaluated and a design based on the compact stellarator (CS) concept has been recently developed by the ARIES team. Nuclear analysis is needed to obtain some key neutronics parameters, such as the neutron wall loading level, tritium breeding ratio, and radiation damage to structural components. These nuclear analyses give guidance and recommendations on radiation protection for the TF magnet, the size of a breeding blanket, and the selection of an optimal shield.

A three-dimensional Monte Carlo analysis is often needed by the MCNP code to generate the neutron wall loading profile and estimate the overall design parameters such as the tritium breeding ratio and energy multiplication. However, the present MCNP code only provides limited geometry modeling capabilities, particularly for complex geometries. A limited number of geometric primitives are difficult to use when constructing complex models. ARIES-CS involves complicated geometries that can not be modeled with the present version of MCNP. If we use an approximation of the actual geometry configuration, the accuracy of Monte Carlo calculation results will be inevitably hurt by an inaccurate geometry model.

To improve the modeling capability of MCNP, we used CAD software as the geometry engine. The radiation transport is directly performed through the CAD geometry. We use the Common Geometry Module (CGM), which is based on ACIS, as the CAD geometry engine. The MCNP/CGM code can perform simulations on any CAD geometry model. Because complex models are usually designed with CAD systems, the CAD model of complicated geometry usually exists before the Monte Carlo simulation. Therefore, we can also save the time of modeling the complicated geometry for the Monte Carlo calculation.

The first application of the CAD-MCNP coupling approach for the ARIES-CS design is to calculate the neutron wall loading distribution (□) in the poloidal and toroidal directions. Variations of the neutron source profile within the plasma boundary have been examined. These include a uniform source across the plasma, a line source at the geometric or magnetic axes, and the actual source profile that peaks at the geometric magnetic axis. It is found that the change to the peak □ due to the neutron source distribution is about 20%.

The results of this analysis will have a major impact on the ARIES-CS design. The poloidal/toroidal □ distribution helps determine the exact size of the shield needed to protect the magnet and thus could solve a potential interference problem that has been identified for the field-period maintenance scheme when the blanket is moved toroidally out for replacement at the end of its service lifetime.

Blanket Design and Optimization Demonstrations of the First Wall/Blanket/Shield Design and Optimization System

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In fusion reactors, the blanket performance and characteristics have a major impact on the reactor performance, size, and economics. The selection and arrangement of the blanket materials, dimensions of the different blanket zones, and different requirements of the selected materials for a satisfactory performance are the main parameters, which define the blanket performance. These parameters translate to a large number of variables and design constraints, which need to be simultaneously considered in the blanket design process. This represents a major design challenge because of the lack of a comprehensive design tool capable of determining all these variables to define the optimum blanket design and satisfying all the design constraints based on the adopted figure of merit and the blanket design criteria.

The blanket design capabilities of the First Wall/Blanket/Shield Design and Optimization System (BSDOS) have been developed to overcome this difficulty and to provide the state-of-the-art research and design tool for performing blanket design analyses. This paper describes some of these capabilities and demonstrates its use. In addition, the use of the optimization capability of the BSDOS can result in a significant blanket performance enhancement and cost saving for the reactor design under consideration. In this paper, examples are presented, which utilize an earlier version of the ITER solid breeder blanket design and a high power density self-cooled lithium blanket design to demonstrate some of the BSDOS blanket design capabilities.

A Self-Cooled Lithium Blanket Concept for the HAPL Conceptual Laser IFE Power Plant

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The High Average Power Laser (HAPL) program is carrying out a coordinated effort to develop Laser Inertial Fusion Energy (Laser IFE) based on lasers, direct drive targets and a dry wall chamber. A primary focus of the study is the development of a tungsten-armored ferritic steel (FS) first wall (FW) which must accommodate the ion and photon threat spectra from the fusion micro-explosion. Only a thin region of the armor (10-100 μm) will experience the highly cyclic photon and ion energy deposition transients. The first wall structure behind the armor as well as the blanket will operate under quasi steady-state thermal conditions, very similar to MFE conditions. This allows the possibility of making full use of information from the large international MFE blanket effort in adapting a blanket for the laser IFE case. Scoping studies of several blanket concepts compatible with the chosen FW protection scheme are ongoing as part of a first phase effort with the goal of converging on the most attractive concept(s) for more detailed integrated studies during a second phase. One of the concepts considered as part of this initial effort is a self-cooled lithium blanket. This is an interesting concept when applied to IFE (as contrasted to MFE) since the absence of magnetic field allows the designer to take advantage of the high heat transfer capability of lithium without the MHD issue.

The geometry of the chamber is near cylindrical. For the example reactor size assumed in the analysis, the radius of the chamber is 6.5 m at the mid-plane and tapers to 2.5 m at the upper and lower extremities. The blanket consists of banks of rectangular tubes (or submodules) arranged vertically extending the whole height of the chamber. The FW of the submodule is 0.35cm thick ferritic steel, which has a 0.1 cm thick tungsten armor layer diffusion bonded to it facing the target. The rectangular tubes vary in width and depth to accommodate the shape of the chamber. Concentric with the outer tube, is another inner tube situated inside the submodule and separated from the outer wall to form an annular channel. The outer wall (including the FW) is cooled with Li which is admitted at the bottom of the blanket, flows through the annular gap to the top and then returns to the bottom at low velocity through the large center channel provided by the second concentric tube. This permits to some extent the decoupling of the lithium outlet temperature from the wall temperature. To facilitate maintenance and accommodate the numerous (~60) beam ports, a number of submodules are lumped together to form a module. The middle submodule in each module has the same width top to bottom, and contains the beam ports. Behind the blanket modules is a ferritic steel vacuum chamber, which will be cooled with He gas. Preliminary neutronics analysis shows that a blanket of at least 47 cm is required in order to make the vacuum chamber a lifetime component. Assuming a 200 dpa limit, the blanket will have a useful lifetime of 10 full power years. The thickness of the vacuum chamber is 50 cm which allows for rewelding at the back. The overall tritium breeding ratio (TBR) is estimated to be 1.1. The blanket is coupled to a Brayton power cycle through a Li/He heat exchanger. Details of the scoping study of this blanket are presented in this paper.

Maintenance Approaches for ARIES-CS Power Core

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The configuration and maintenance approaches for a power plant based on a compact stellarator are very different than for one based a tokamak. Compared to a tokamak, the replacement of the power core in a stellarator is considerably more challenging because the access to the blanket is strongly limited by the shape of the modular coils. The engineering effort during the first phase of the ARIES-CS study has focused on scoping out different compact stellarator design configurations and maintenance schemes to determine the key issues and better understand the parametric design windows and the engineering constraints. Three possible maintenance approaches for the ARIES-CS compact stellarator have been considered during this scoping phase:

1. Field-period based replacement including the disassembly of the coil system.
2. Modular replacement approach through maintenance ports arranged between each adjacent pair of modular coils.
3. Modular replacement through a small number of designated maintenance ports using articulated booms.

Individually, these different schemes impose different constraints on the physics configuration of the compact stellarator; for example, the field-period based replacement scheme tends to be more suited for configurations with three (e.g. NCSX) or more field periods and the modular replacement approach through maintenance ports arranged between each adjacent pair of modular coils requires adequate port space between the coils and tends to be better suited for a two-field period configuration. However, as a whole, these choices of maintenance schemes provide a sufficiently broad range of possibilities to accommodate the physics optimization on the machine configuration and size (including the number of coils and number of field periods).

The key issues associated with the three maintenance approaches have been identified and addressed, and the layout of coil system, coil supporting structure, cryostats, vacuum vessel and maintenance ports have been conceptually defined. These are summarized in this paper. Blanket concepts compatible with the three maintenance approaches have been investigated, and will be reported in separate papers. State-of-art CAD tools have been used in this study to verify clearances for blanket movements during maintenance operation and to determine the size of the blanket modules based on the available space for module removal.

Manufacturing Concepts for an IFE Power Plant Using Z-Pinch Technology*

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The Z-Pinch Power Plant uses the results from Sandia National Laboratories' Z accelerator in a power plant application to generate energy pulses using inertial confinement fusion. A collaborative project has been initiated by Sandia to investigate the scientific principles of a power generation system. Research is underway to investigate the use of recyclable transmission lines to directly connect the wire array and the hohlraum directly to the pulsed power driver. The resulting power plant will require an intense on-site manufacturing system to rebuild the transmission lines, wire arrays and hohlraums at a rate of 0.1 Hz per power unit. By recycling virtually all of the materials, the system is expected to be economically competitive with other power generation technologies. Current research is investigating the available approaches to manufacturing and determining the cost effectiveness of the alternatives. This paper examines the various options available for manufacturing and development requirements leading to a Proof-of-Principle experiment to demonstrate the technology.

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National Compact Stellarator Experiment (NCSX) Vacuum Vessel Manufacture

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The National Compact Stellarator Experiment (NCSX) is the first of a new class of stellarators known as "compact stellarators." Stellarators are a class of magnetic fusion confinement devices characterized by three-dimensional magnetic fields and plasma shapes and are the best-developed class of magnetic fusion devices after the tokamak. The stellarator concept has greatly advanced since its invention by Dr. Lyman Spitzer, the founding director of the Princeton Plasma Physics Laboratory (PPPL), during the 1950's. A traditional stellarator uses only external magnetic fields to shape and confine the plasma. The differentiating feature of a compact stellarator is the use of plasma current in combination with external fields to accomplish plasma shaping and confinement. This combination permits a more compact stellarator design.

NCSX requires a highly shaped, three-period vacuum vessel, which means the geometry repeats every 120º. Stellarator symmetry also causes the geometry to be mirrored every 60º so that the top and bottom sections of the first (0º to 60º) segment can be flipped over and serve as the corresponding sections of the adjacent (60º to 120º) segment. The vessel will be constructed in full field periods and joined together at welded joints. Numerous ports are provided for heating, diagnostics, and maintenance access. Several port sizes and shapes are used to best utilize the limited access between modular coils.

Each 120º segment is called the Vacuum Vessel Sub-Assembly (VVSA) and includes associated port extensions and spacer assemblies, which are critical components for NCSX. The VVSA's, fabricated of N06625 Inconel material, are toroidal (donut-shaped) in major diameter, but highly shaped in the poloidal (short) direction. Three (3) 120º VVSA's comprise the complete NCSX vacuum vessel. Each VVSA includes a field period assembly (the basic vacuum vessel shell), associated port extensions (including blank off flanges, seals, and fasteners) and one connecting spacer assembly.

Fabrication is a significant challenge, since the vessel has a contour closely conforming to the plasma on the inboard side. Inconel 625 was selected over stainless steel primarily because of its low permeability (both in the parent and weld material) and high electrical resistivity. The vessel shell is formed by pressing plate sections, then welding them together to form the finished shape. Segmentation of the vessel is driven by assembly requirements and inherent fabrication limitations. Fabrication by pressing requires the panel sections to be removable from the tooling dies. This requirement must mesh with the desire for half-period segments. The result is that the number and geometry of poloidal segments is dictated by the die contour.

The form tolerance of the vessel must be very accurate in the inboard region, with a tolerance of +/- 0.188 inches to provide adequate clearance to both the coils and the plasma. These tolerances must be held after the vessel is completely welded and assembled, so intermediate heat treatments during fabrication may be necessary.

Experiments to Improve Power Conversion Paramters in a TWDEC Simulator

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Traveling Wave Direct Energy Converter (TWDEC) is used for efficient energy conversion of high energy protons produced in an advanced fusion (D-^{3}He) reactor. The authors are continuing experimental research to realize TWDECs.

In a TWDEC, a fast proton beam is introduced in a modulator where the beam is velocity-modulated. The beam is density-modulated in the downstream where a decelerator is settled and the modulation frequency is induced on aligned electrodes. The electrodes are connected to an external circuit, and current flows in the circuit forms a traveling voltage on the electrodes, and thus, a deceleration field between the electrodes. The beam is decelerated by this field. In order to examine a desirable structure of the decelerator, the authors are continuing simulation experiments in two ways. One is for formation of deceleration fields, and we call this working mode as a passive decelerator. The other is for deceleration of an ion beam, and we call as an active decelerator. In this paper, we will present improved results of each working mode.

As for the passive decelerator, it is required to create a strong deceleration field. For this purpose, we introduced a new type of external circuit, which we called as loop type. We had used a line type external circuit before, the ends of which were terminated by matched resisters and an induced power was consumed by the resisters. In the new circuit, both ends are connected to each other and there are no resisters. The induced power is consumed by just small resistive components of inductors, thus voltages on electrodes are expected to be higher.

We actually constructed a loop type external circuit, and obtained results that the excited voltage was higher in an order of magnitude. The results also indicated a problem that higher harmonic frequencies were excited as a Q-value of the circuit was high. We are continuing works to suppress them.

As for the active decelerator, we seek for a desirable structure of decelerator electrodes. The beam current of our TWDEC simulator is rather small, so the deceleration field created by the passive decelerator is weak. We have performed experiments creating a deceleration field by using an external power source.

We had examined variation of deceleration efficiency, which directly equaled to recovery efficiency of ion energy. We obtained a conclusion: the deceleration efficiency saturated when the phase velocity of traveling wave was constant, but it extended beyond the saturation level when we varied the phase velocity to match with the decelerated beam velocity. We have proved this effect in principle by adjusting the phase difference of voltage between electrodes. In a real TWDEC, this is realized by adjusting distances between electrodes. We are now performing experiments to adopt this scheme to obtain the best deceleration efficiency. The details will be presented in the meeting.

Experimental Simulation on Particle Discrimination for Direct Energy Conversion

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A direct energy conversion system designed for D-^{3}He fusion reactor based on a field reversed configuration employs a Venetian blind type converter for thermal ions to produce DC power and a traveling wave type converter for fusion protons to produce RF power.[1)] It is therefore necessary to separate, discriminate, and guide three major particle components; electrons, thermal ions, and high-energy protons.

For this purpose, proposed is a cusp magnetic field, in which the electrons are deflected and guided along the field line to the line cusp, the thermal ions are less deflected but flow into off-axis region, and the protons pass through the point cusp maintaining on-axis orbit.

We have constructed an experimental device, which consists of a low-energy plasma or ion source, a guide field section, and a cusp magnetic field section. The device is capable of changing the curvature of the magnetic fields from normal cusp to slanted cusp field. We inject the plasma or ion beam into the slanted cusp field to simulate the separation of electrons and ions or the discrimination of ions with different energies.

The cusp field is created by two magnetic coils A and B, with currents I_A and I_B, respectively. A plasma beam is injected into the point cusp of the coil A side. We measure the ion and electron fluxes with changing I_B for a fixed I_A = 30 A. We define the transmission ratio of the particles as the ratio of the flux at the point cusp of the coil B side to that at the entrance. When the cusp field is formed, most of the electron flux flows into the line cusp and most of the ion flux into the point cusp, yielding the transmission ratio of electrons to a small value and that of ion close to 1. The best separation of electrons and ions is obtained when $I_B / I_A = 1.4$ and the location of the electron collector is at a half way to the line cusp exit. The transmission ratio of electrons in this case is as low as 0.05.

By using the ion source, we simulate the energy discrimination of the thermal ions from the protons in a reactor. The ratio of the ion current at the point cusp to that at the line cusp is measured versus the energy of the incident ion beam. The ratio increases steeply and then saturates as the energy of incident ions is increased from 0.1 to 1 keV, showing that higher energy ions preferentially flow into point cusp.

These experiments will be performed for a variety of energy range to establish a scaling law of particle discrimination in the cusp field for the direct energy conversion.

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Optimized Helium-Brayton Power Conversion for Fusion Energy Systems

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This paper will present an overview and a few point designs for multiple-reheat helium Brayton cycle power conversion systems using molten salts (or liquid metals or direct helium cooling). All designs are derived from the General Atomics GT-MHR power conversion unit (PCU). The important role of compact, offset fin heat exchangers for heat transfer to the power cycle helium, and the potential for these to be fabricated from carbon-coated composite materials that would have lower potential for fouling. Specific links will be made to the ITER TBM and laser IFE blanket design, and to Z-Pinch/HIF thick-liquid IFE.

The GT-MHR PCU is currently the only closed helium cycle system that has undergone detailed engineering design analysis, and that has turbomachinery which is sufficiently large to extrapolate to a >1000 MW(e) molten salt coolant gas cycle (MCGC) power conversion system. Analysis shows that, with relatively small engineering modifications, multiple GT-MHR PCU's can be connected together to create a MCGC power conversion system in the >1000 MW(e) class. The resulting power conversion system is quite compact, and results in what is likely the minimum helium duct volume possible for a multiple-reheat system. To realize this, compact offset fin plate type salt-to-helium heat exchangers (power densities from 10 to 120 MW/m^3) are needed. Compact plate heat exchangers are already commonly used for heat transfer at lower temperatures. Of great interest for fusions is the potential to fabricate compact plate type heat exchangers that would provide very high surface area to volume ratios and very small fluid inventories while operating at high temperatures. Both metal and non-metal heat exchangers are being investigated for high-temperature, gas-cooled reactors for temperatures to 1000ºC. Recent high temperature heat exchanger study for nuclear hydrogen production has suggested that carbon-coated composite materials such liquid silicon infiltrated chopped fiber carbon-carbon preformed material potentially could be used to fabricate plate fin heat exchangers that would have lower potential for fouling. Optimization and cost models for the power conversion systems applicable to fusion power plants could be used to select optimum parameters such as system pressure, numbers of reheat and intercooling, and other important parameters. This paper summarizes a few power conversion system point designs for three options of ITER US DEMO blanket designs. Among three blanket design options, self-cooled FLiNaBe salt option potentially can achieve the best thermal efficiency through MCGC. For dual-coolant options, dual cycles, helium direct cycle plus MCGC cycle can combine to achieve a good thermal efficiency.

Radiation Induced Conductivity of Proton Conductive Ceramics Under 14 MeV Fast Neutron Irradiation

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Proton conductive ceramics are applied as high temperature protonic conductors in electrochemical devices as well as tritium breeding materials in fusion devices. Further applications for tritium monitors during reactor operation or tritium collectors from vacuum pumping gas and tritium compounds (oxidized tritium, tritiated water vapor) are expected, because hydrogen extraction performances from hydrogen molecular, methane and water have been confirmed and evaluated by using the closed-end type proton conductive ceramic tubes.

In the present study, electronic and protonic conductions of typical proton conducting oxide ceramics ($SrCe_{0.95}Yb_{0.05}O_{3-\delta}$) which were perovskite-type were investigated at room temperature and 373 K under 14.1 MeV fast neutron irradiation at the facility of Fast Neutron Source (FNS) in Tokai Research Establishment of Japan Atomic Energy Research Institute (JAERI). The average fast neutron flux was about $3.0\text{x}10^{12}$ n/m^2 s. The accompanying gamma-ray dose rate was less than 1 μGy/s. The relation between current and voltage at room temperature and 373 K during the irradiation at several fast neutron fluences was observed and compared with those without radiation. The radiation-induced conductivity (RIC), calculated using Ohm's law from the measured current and applied voltage, decreased with increase of the irradiation fluence and became nearly a constant at the neutron fluence of about $1.0\text{x}10^{17}$ n/m^2, due to radiation-induced defects and annihilation of sub-bands, produced oxygen vacancies and impurities such as hydrogen, in the band gap. The radiation damage did not recover at room temperature and 343 K after the irradiation. Then, the RIC at room temperature only slightly incresed with further irradiation. This increase is probably caused by the electronic and protonic excitations by high-energy neutrons.

Effect of Initial Heat Treatment on DBTT of F82H Steel Irradiated by Neutrons

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Reduced-activation ferritic/martensitic steels are candidate materials for the blanket structure of fusion reactors. Radiation-hardening of 8-9%Cr martensitic steels irradiated by neutrons occurrs mainly at irradiation temperatures lower than about 400^{o}C, and it increases with decreasing irradiation temperature down to 250^{o}C. The shift of DBTT (ductile-brittle transition temperature) also increases with decreasing irradiation temperature, and the shift increases largely for irradiation at 250^{o}C. Several researchers reported that the increase of yield strength and the shift of DBTT were different in several martensitic steels, such as F82H, JLF-1, JLF-1B, ORNL 9Cr-2WVTa, OPTIFER Ia, II, MANET II and Mod.9Cr-1Mo, which had different concentrations in some elements and were tempered at different temperatures. The effects of the normalizing and tempering of heat treatment on tensile and impact behavior in martensitic steels before irradiation were reported by L. Schafer and P. Gondi. However, the mechanisms of the changes of yield strength and DBTT due to irradiation in these martensitic steels are not clear, and it is necessary to reveal the effects of heat treatment and impurities on them. The optimum heat treatment will be required to improve resistances to radiation hardening and embrittlement. In this study, the dependence of impact properties on tempering time and temperature has been examined for martensitic steel F82H (Fe-8Cr-2W-0.2V-0.04Ta-0.1C) irradiated by neutrons.

The dependence of DBTT on tempering time and temperature was examined for a martensitic steel F82H irradiated at 150 and 250^{o}C to a neutron dose of 1.9 dpa in the Japan Materials Testing Reactor. Miniaturized Charpy V-notched (CVN) impact (3.3 mm x 3.3 mm x 23.6 mm) specimens were fabricated. The heat treatment was performed at 750 and 780^{o}C for 0.5 h after the normalizing at 1040^{o}C for 0.5 h. The tempering time at 750^{o}C was varied from 0.5 to 10 h. Charpy impact testing was carried out in the hot cell of the JMTR of JAERI, and the absorbed energy was measured as a function of temperature. After the testing, the fracture surfaces were observed. The DBTT of F82H steels heat-treated with different tempering conditions depended on the tempering conditions before the irradiation. After irradiation, the DBTT of F82H steels irradiated at 250^{o}C to 1.9 dpa was ranged from –23 to 25^{o}C, and the DBTT of F82H steels irradiated at 150^{o}C to 1.9 dpa was ranged from 0 to 15^{o}C. The DBTT of the F82H steel depended on temperature and time of tempering. In this study, the DBTT of pure iron and F82H+2Ni steel irradiated by neutrons was also examined.

Tungsten Coating on Low Activation Vanadium Alloy by Plasma Spray Process

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Tungsten metal and low activation vanadium alloys are promising candidates for the plasma facing material and the structural materials, respectively, in the first wall of fusion blanket system. Tungsten has a high melting point, such as 3650 K, and low sputtering rate in fusion plasma environment. Low activation vanadium alloys have demonstrated high mechanical strength up to 1000 K, high resistance to neutron irradiation and manufacturing feasibility, such as large scale melting, parts fabrication and welding. In the present study, tungsten coatings on the low activation vanadium alloy were fabricated by plasma spray process, which is practical for large area coating because of its high coating rate. The tungsten coating, interface and vanadium alloy substrate after coating are characterized. Engineering issues, bonding property and behavior of the constituent element and pick-up impurities were discussed.

The vanadium alloy substrate was a 2 mm-thick plate of NIFS-HEAT-2, which is the reference high purity V-4Cr-4Ti alloy. Tungsten powder of 99.9 % grade and in 325 mesh size was used as coating materials for plasma spray process. In the process, the vanadium alloy substrate surface was blustered and cleaned by argon arc sputtering. The tungsten powder was carried by argon gas to the plasma jet of argon and hydrogen mixture. The jet sprayed the substrate with the melted tungsten particle. The power of the plasma gun and scanning speed of the gun was 45-46 kW and 75 mm / sec, respectively. Initial and final surface temperature at coating was 620-720 K and 920-1070 K, respectively. The plasma spray chamber was filled with argon gas of about 10^4 Pa in pressure. The resulting thickness of the tungsten coatings was about 0.5 mm after 10 scans.

The tungsten coating side was convexly curved because of the difference in the thermal expansion coefficient of the tungsten and vanadium alloys, which is 4.6 and 9.8 $\times 10^{-6}$ K^{-1}, respectively, for the average in RT-773 K. The deformation did not induce peeling or cracking of the coating. Hardness of the tungsten coating was varied from 350-640 Hv with the load of 25 g, while that of the vanadium alloy substrate was 210-250 Hv. The large scattering of the hardness in the tungsten coating region was considered as a result of inhomogeneous solidification and rapid cooling rate of the tungsten coating. At the vanadium alloy substrate, oxygen contamination was detected around the interface to the tungsten coating, however an accompanied hardening and its depth from the interface were limited as about 10 Hv and 100 μm, respectively, which was expected not to induce significant change in mechanical property of vanadium alloy substrate. In the present study, the direct deposition and coating fabrication of tungsten on the vanadium alloy was successfully demonstrated.

Hydrogen Embrittlement Susceptibility of Conventional and Reduced Activation 9Cr-Steels

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Reduced activation ferritic/martensitic steels (RAF/MS) for DEMO structural components have been derived from the conventional 9-12Cr alloys by elemental substitution of Mo, Ni and Nb with W, V and Ta respectively for mitigating the high level waste issue. Further compositional adjustments for the purpose of improving their mechanical performances and irradiation damage resistance have led to the development of recent modified versions which actually are under characterisation within the frame of International Material Programmes.

One among other uncertainties to be solved for a safe and effective usage of RAF/MS concerns their potential susceptibility to embrittlement phenomena, due to their simultaneous exposure to thermal stresses and hydrogen generated from various sources in fusion reactor working scenarios. The nature and entity of hydrogen induced damage are well known to strongly depend on material factors, especially microstructural trap types, density and distribution, and thus may considerably differ from steel-to-steel of the same family.

The objective of this work was to compare the intrinsic propensity to hydrogen degradation of the conventional 9CrMoVNb steel, T91, to those of two 9CrWVTa RAF/MS: the experimental heat, VS3104, and Eurofer'97, which is the current reference alloy of the European strategy. For this purpose, constant extension rate tests were performed on tensile specimens pre-saturated with various amounts of hydrogen by electrochemical galvanostatic charging, and maintained at saturation during mechanical testing to avoid any hydrogen loss. Charged versus uncharged reductions of specimen areas at rupture were taken as the most suitable ductility parameters for quantifying the magnitude of hydrogen embrittlement. Immediately after testing, the specimens were subjected to a linearly increasing heating ramp in a furnace for hydrogen desorption and quantitative analysis by differential conductivity measurements. The very selective character of the extraction process gave an indirect indication of the number and strengths of the effective material defects for hydrogen trapping, while conclusions about their possible nature were drawn on the basis of microstructural assessments. Predictions concerning the implication of specific traps to the cracking process induced by hydrogen could be made with the support of scanning electron microscope fractographic analysis.

Synergistic Influence of Displacement Damage and Helium/dpa on Microstructural Evolution and Radiation-Induced Hardening of Reduced Activation Ferritic/Martensitic Steel

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Reduced activation ferritic/martensitic steels, RAFs, are leading candidates for blanket and first wall of fusion reactors where effects of displacement damage and helium production are important subjects to be investigated. To obtain systematic and accurate information of microstructural response under fusion environment, dual-ion irradiation method was applied. In order to estimate the microstructural response under fusion neutron irradiation environment, ion-beam irradiation was carried out with helium and metallic self ions. The objective of this work is to clarify correlation between irradiation hardening and microstructural evolutions in RAFs under ion irradiation to high fluence.

JLF-1 (9Cr-2W-V, Ta) steel was irradiated up to 60 dpa at 693, 743 K. Single ion irradiation was performed with 6.4 MeV Fe^{+3}. Energy degraded 1.0 MeV He^{+} was simultaneously irradiated for dual ion irradiation. The damage rate and helium injection rate were 1.0×10^{-3} dpa/s and 15 $\times 10^{-3}$ appmHe/s. As the post-irradiation examination, transmission electron microscopy (TEM) and nano-indentation were carried out.

Microstructural evolution was characterized for irradiation conditions where significant changes in micro-hardness in RAFs were found. At 693 K, irradiation hardening was not decreased under single-ion irradiation. Under dual-ion irradiation, the dislocation structure consisted of loops and network was detected. Here, an important role of dislocation evolution in irradiation hardening is confirmed. Irradiation hardening in dual-ion irradiation at 693 K was larger compared to that of single-ion irradiation. At 743 K, void cavity structure was observed under dual-ion irradiation where the contribution of void structure on hardening was not so significant.

JLF-1 under single-ion irradiation exhibited excellent irradiation resistance, although irradiation hardening and swelling were depended for the case of dual-ion irradiation. However microstructural evolutions could be detected mostly on lath structure. Contribution of dislocation structures, distributions of precipitate and others on swelling and hardening will be presented in detail.

Corrosion Behavior of Insulator Coatings for Fusion Reactor Lithium Blankets

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A liquid lithium blanket is an attractive system because of its high effectiveness, low irradiation, continuous operation and its compact system. However, there is a serious problem called MHD pressure drop, which is Lorentz force opposite direction to the lithium flow. To solve this problem, it is suggested to construct insulate coatings on inner pipe walls. The coatings must have high resistivity to reduce MHD pressure drop, and also have high chemical stability in liquid lithium, which has high reduction activity, at high temperature. Several ceramic materials were suggested from the point of chemical stability and investigated with bulk materials by sintering in liquid lithium, AlN, Y_2O_3 and Er_2O_3 are thought as good candidate materials. The purpose of the present study is to characterize corrosion behavior of the coatings of the three candidate ceramics fabricated by RF sputtering.

The coatings were sintered in 300-500 ^oC liquid lithium, and after the sintering the specimens were sintered in water to remove lithium remaining on the specimens. In the case of Y_2O_3 and Er_2O_3 coatings, the samples sintered at 300 ^oC were damaged little, while the coatings almost disappeared after sintering at 500 ^oC. Observation by SEM showed that there were little pits on the surfaces of the specimens sintered at 300-400 ^oC, and there were little fragments of the coatings remaining on the specimens sintered at 500 ^oC. This suggests that there were some pits on the coatings, and the coatings were broken and peeled off from the pits. There were already same kinds of pits on the coatings after annealing at 300-500 ^oC. In addition to the sintering in water, vacuum distillation method was applied to remove the remaining lithium, for the samples after sintering in liquid lithium. Although there were many pits on the surface of the coatings, there was almost no destruction or separation. It is considered that the pits were formed by the heating, not by the reaction with liquid lithium, and thus the coatings were stable in liquid lithium. On the other hand, AlN coatings had no pits after annealing; however, the coatings almost disappeared even after sintering at 300 ^oC, with only small fragments remaining. It is considered that destruction and separation of the coatings also happened in the case of AlN coatings.

Cavity Swelling Behavior in SiC/SiC under Charged Particle Irradiation

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Silicon carbide (SiC) and its composites (SiC/SiC) are attractive structural materials for fusion reactor because of their superior mechanical properties at high temperature and high irradiation resistance. On their performance under fusion nuclear environment, void swelling is one of the important issues to be clarified. As to microstructural evolution, authors have reported that at high dose helium cavities were formed on grain boundaries in both matrix and fiber at 1273 and 1673K. However, the effects of displacement damages and of helium production on microstructural evolutions of matrix, fiber and interface have not been sufficiently investigated yet. The objective of this study is to analyze microstructural evolution in SiC/SiC composites under ion irradiation at high temperature, with the special emphasis on behavior of cavities in SiC/SiC.

The materials used were Tyranno-SA^{TM} fiber reinforced SiC/SiC composites, fabricated by chemical vapor infiltration (CVI) method. The ion irradiation was carried out at Dual-Beam Irradiation Experimental Test (DuET) Facility in Institute of Advanced Energy, Kyoto University. In order to induce displacement damages, 5.1MeV Si^{2+} ions were irradiated (single-ion irradiation). And in order to simulate (n, α) reaction, an additional beam of energy-degraded 1.0MeV He^{+} ions was simultaneously implanted (dual-ion irradiation). Displacement damage level was up to 100dpa, irradiation temperature was up to 1673K, He/dpa ratio was 0 or 60appmHe/dpa, respectively. For a microstructural investigation by cross-sectional transmission electron microscopy (XTEM), the irradiated samples were subjected to a thin foil processing using a focused ion beam (FIB) device.

In single-ion irradiation, no void was detected at 1273 and 1673K, 10dpa. But many voids were observed at 1673K, 100dpa. In dual-ion irradiation, three kinds of cavities were observed at 1673K, 100dpa: The first one is helium bubbles (approximately d=5nm) formed densely on (111) faulted planes in the fiber and matrix. The second one is voids (approximately d=20nm) formed on grain boundaries in the fiber and matrix. The third one is large cavities (approximately d=100nm) observed on grain boundaries only in the fiber. The third types were pre-existing pores as intrinsic defects. There are helium bubbles observed densely around such large cavities. As total dose dependence, reduction of helium bubble number density, void growth and localization were detected. The nucleation sites of cavities, grain size and other factors affecting microstructural evolution under single or dual-ion irradiation will be discussed.

An Innovative Solid Breeder Material for Fusion Applications

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A new form of solid breeder morphology is proposed, which has the potential of increased breeding ratio, long term structural reliability, and enhanced operational control compared to conventional approaches, such as packed bed). Breeding ratios are dictated by breeder material density, while tritium release depends on diffusion path length to free surfaces. Packed bed concepts for solid breeder materials have offered a venue for large free surface areas to accommodate high release rates. However, the maximum breeder material packing factors are limited to about 65% for single size pebbles, which can be raised only with the addition of very small (e.g., 80-150 μm) pebbles.

The pebble bed configuration introduces several operational limits due to thermo-mechanical uncertainties caused by pebble bed wall interaction, potential sintering and subsequent macro-cracking, and a low pebble bed thermal conductivity, all of which result in small characteristic bed dimensions and operation windows. Consequently, long term and reliable performance of pebble beds remains a critical and complex issue for fusion reactors.

We suggest here for the first time the use of a solid ceramic “breeder foam” material, which offers several and significant advantages over the use of pebble bed configurations. In recent years the development of refractory and ceramic foams has matured to a degree that detailed thermo-mechanical properties can be micro-engineered into these advanced structures. For fusion breeder applications, “breeder foams” offer several advantages over pebble beds: (1) increased breeder material densities (>80%), (2) higher thermal conductivities (fully interconnected structure instead of point contacts between pebbles), (3) bonded contacts to cooling structures (instead of point contacts between pebbles and wall), (4) no major configurational changes between beginning-of-life and end-of life (such as sintering in pebble beds); and (5) structural integrity because foams are freestanding and self-supporting structures with significant thermo-mechanical flexibility.

Thermo-mechanical properties of oxide ceramic foams are discussed, and manufacturing techniques for lithium-based oxide foams are presented based on current day ceramic oxide foam manufacturing processes. Also, high-temperature foam-to-solid-surface bonding techniques are presented.

Helium Retention of Ion-irradiated and annealed Tungsten Foils

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In an inertial fusion energy (IFE) chamber, the tungsten armor on the first wall will be subjected to intense doses of ions. The most deleterious ions are helium in the energy range of 0.1-1 MeV. He ion irradiation damage creates vacancies within W that trap He and act as nucleation sites for He bubble growth. He trapping and bubble growth increases with dose and irradiation temperature due to increasing defect density and thermal mobility. At room temperature growth of He bubbles beneath the surface causes blistering at ~3 x 10^{21}/m^2 and surface exfoliation at ~10^{22}/m^2. These critical doses increase with ion energy (in~1-2 MeV range), and are also dependent on annealing temperature as well as the annealing sequence. However, there is insufficient information on the effect due to MeV He ions and elevated temperature exposures and anneals. To understand the helium retention characteristics and helium bubble distribution of tungsten, a ^{3}He(d,p)^{4}He nuclear reaction analysis technique and transmission electron microscopy (TEM) have been performed for two forms of tungsten: single crystal and polycrystalline, implanted up to 1x10^{19} ^{3}He/m^2 and annealed to 2000°C.

Samples were implanted at 850°C, flash-heated at 2000°C, and analyzed by ^{3}He(d, p)^{4}He nuclear reaction with 780 keV deuterons. Surface blistering was observed for doses greater than 10^{21} He/m^2. Single crystal and polycrystalline tungsten samples implanted with 1 x 10^{19} He/m^2 at 850°C exhibited similar helium retention characteristics. A flash anneal at 2000°C had no effect on the retention of helium. This dose was low enough to avoid surface blistering, but high enough to result in strong helium trapping and bubble growth. Implantation and flash-heating in cycles indicated that retention strongly depends on the He dose per cycle and tungsten microstructure. When 10^{19} He/m^2 was implanted into single crystal tungsten in 1000 cycles (10^{16} He/m^2 per cycle), the observed helium yield dropped to ~5 % compared to ~30 % for polycrystalline tungsten. Considering the experimental results presented, the first wall of an IFE fusion reactor will potentially suffer from significant damage due to high fluences of helium ions and intense temperatures. Helium trapping and bubble formation just below the surface of the first wall material will result in surface blistering and exfoliation at critical helium doses. The data also suggest preference of single crystal over polycrystalline tungsten due to less retention compared to polycrystalline.

Elasto-plastic Dislocation-based Constitutive Modeling within Full Scale 3-D Thermo-mechanical FEM Analysis of an ITER-TBM

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For the near term, researchers must rely on computer simulation tools to study the behavior of fusion components using material models and finite element analysis. The Virtual international Structural Test Assemble (VISTA) is an international project aimed at establishing an effective interface between material modeling and component design activities. The objective is to combine a wide range of models including constitutive and damage laws, finite element models, geometrical configurations and loading conditions, to perform "virtual experiments" over a wide range of conditions, to carry out sensitivity studies and to evaluate a range of potential interactions and failure paths.

Materials modeling provide predictive relations between the microstructure of the material and its macroscopic mechanical properties. A dislocation-based phenomenological equation of the elasto-plastic constitutive behavior for steels as a function of temperature and neutron irradiation damage (dpa) is developed. The model is applied to and calibrated with experimental stress strain data of low activation ferritic steels, such as HT9 and F82H. Model calibration is based on a 3-dimensional (3-D) finite element analysis of a tensile stress bar.

Aside from various material propertie's the failure paths of a component are highly dependent on geometric features, loading conditions, and interaction between loading and damage. As a first step of developing the VISTA concept it is necessary to develop detailed full-scale 3-D models of entire components in order to capture the effects of 3-D geometric features and to be able to apply realistic boundary conditions. To this end the entire component must be modeled. We present here a dislocation-based material model, which is employed in a detailed full-scale 3-D FEM of the entire EU HCPB TBM. A full thermo-mechanical analysis is performed on the ITER test module using the dislocation-based phenomenological creep model. Realistic boundary conditions could be applied because the entire TBM is modeled. Steady state and transient thermo-mechanical response of the EU HCPB ITER-TBM are reported and compared with results based on purely elastic analysis.

Upper Critical Field Improvement in MgB_2 by Mechanical Alloying

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The new superconductor magnesium diboride (MgB_2) shows great promise for fusion applications because it has a higher Tc (39K) than Nb_3Sn, does not suffer from grain boundary weak links due to grain orientation, is potentially inexpensive, and can be made as round multifilament wires. The main challenge for MgB_2 research has been to improve upon the relatively modest upper critical field of the pure material (generally <15T at 4.2K). Recent studies of magnesium diboride thin films have obtained $H_{c2}(0)^{||}$(H parallel to the Mg and B planes) approaching 70T and $H_{c2}(0)^{\perp}$ about 40T in carbon-doped samples with anomalous c-axis lattice parameters [1]. Bulk, untextured carbon-doped samples fabricated by a CVD method had upper critical fields in excess of 30T at 4.2K [2], about a factor of two lower. Collectively these data show that alloyed MgB_2 can exceed the performance envelope of any Nb-base superconductor at any temperature or field. Our present work involves the synthesis of alloyed MgB_2 powder for high-field wires. We have found that high energy milling of magnesium diboride pre-reacted powder can render the material largely amorphous through extreme mechanical deformation and that it is suitable for mechanically alloying MgB_2 with dopants including carbon. We have found that the slope dH_{c2}/dT is enhanced from 0.51 T/K to 1.08 T/K by milling in the presence of C, consistent with $H_{c2}(0)^{||}$of 33 T. Detailed studies of the C and process dependence of the superconducting properties and the recrystallization kinetics are underway and will be reported at the meeting.

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[2] Wilke et al. Cond mat 0312235 (2003)

Jointing Performance in HTc SC Tape for remountable magnet system

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A concept of the remountable superconducting magnet using HTc SC tapes has been proposed for the future design of the fusion power plant [1] to reduce both costs of the construction the maintenance. The HTc superconductors are used in this concept because their robustness against a heat generation at the jointing parts allows a direct jointing of the magnet. The butt jointing method of HTc superconducting magnet [2] and investigations on the butt jointing of Bi-2223 HTc superconducting tapes have been performed [1,3]. In the previous study [1], dependence of the jointing resistance on the transport current was observed in the experiment that a single layered HTc superconducting tape is jointed with another one.

In this study, therefore, a numerical analysis was performed to investigate the reason this dependence of jointing resistance occurs in the experiment.

And secondly, the experiment to show the performance of butt jointing for a cable was performed by using the cable laminating ten superconducting tapes. The contact surface of laminated superconducting cable was cut down by angle of 45 degree inclined from the tapes.

Through this study, we obtained the following results;.

1. From the numerical analysis where the decrease of the critical current density near the contact surface is assumed, the results of the jointing resistance in case of 0.4 mm had good agreements with the previous experimental results.
2. The case of 0.4 mm notched surface was unrealistic because it was much larger than the actual shape of the contact surface, so that there are possibilities that the 0.4 mm was a scale of the degraded filament.
3. The relationship between compressive stress and the critical current of the laminated cable was confirmed. And it can be predicted that there happened the degradation in the superconducting filaments by increasing the contact stress.
4. The optimum stress, which gives the minimum jointing resistance, therefore, exists due to tradeoff between the reduction of jointing resistance and degradation of SC performance in the filaments.
5. The optimum stress obtained by the experiment was lower than the predicted value through other experiments. The reason will be clarified in the full paper.

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Advanced Options for Modular Stellarator Magnets

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There have been investigations of stellarator magnets that incorporate conventional magnet construction with state-of-the-art materials. The purpose of this paper is to investigate the options for stellarator magnets that utilize extrapolation of present day technology, utilizing high performance superconductors, structure and insulation, as well as aggressive design approaches.

Evaluation of the use of high temperature superconducting materials, in particular highly anisotropic YBCO, on the magnet thickness, shield and cooling requirements, is performed. In addition to YBCO, other high Tc materials will be discussed, including BSCCO 2212 and MgB_2. The status of radiation damage on the performance of the high Tc superconductor is also described. This has impact on the required shielding for the survivability of the superconductor and insulator, as well as the cooling requirements.

The impact of magnet protection on high Tc designs will be discussed. As opposed to low temperature superconductors, it will be difficult to measure the occurrence of quench due to the very low speed of propagation of normal zones. Options for protection and implications for the magnet thickness will be discussed.

Conventional react and wind techniques present difficult obstacles for magnet construction with high performance low Tc superconductors because of the complex nature of the magnets. NbTi has good mechanical properties, but it is limited to low fields unless it is operates temperatures lower than 4 K.. Options of using high Tc and intermediate Tc superconductors capable of react and wind will be presented. Wind and react techniques will be very difficult to implement.

Monolithic windings, using high Tc superconductors, will be presented as an aggressive, speculative means of fabricating the magnets. It offers the potential advantage of ease of manufacturing using rapid prototyping techniques and incorporation of a ceramic insulation during the construction process. In such monolithic magnets the superconductor is rigidly supported, minimizing the required space for structural support in the region where the conductor lies and thus increasing the local current density.

Magnet design options that decrease the coil centroid to plasma distance will be explored. The impact of conductor, structure and overall arrangement choices on this distance will be discussed.

Sn content and alloying effects in ITER Nb_3Sn strand

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The optimization of ITER Nb_3Sn bronze-processed composite strand has been accomplished through both advancements in mechanical drawing of the strand to fine filament sizes and progress in optimizing the quality of the Nb_3Sn superconducting layer. Despite the advancements, further improvement of the Nb_3Sn layer would be desirable in terms of increasing the performance envelope (and thereby the operating margin) of the magnets in both field and temperature. This can be done by (1) increasing the H(T) superconducting envelope, and/or (2) reducing the H(T) superconducting transition breadth, which is inherently large due to the presence of Sn gradients imposed by the diffusional nature of the heat treatment reaction. In recent DC high-field measurements of the resistive superconducting transition down to 1.4 Kelvin (K), a Ta-doped and a Ti-doped ITER Nb_3Sn conductor are shown to have almost identical "best bit" upper critical field (H_{c2}) values at 0 K, but significantly different transition widths. Both conductors have a maximum $\mu_0 H_{c2}(0) = 29.5\pm0.1$ Tesla (T), but the Ta-doped wire has a 0 K H_{c2} distribution of 2.8 T, while the Ti-doped wire has a 0 K H_{c2} distribution of 2.0 T. We explain this result in terms of the Sn supply and Sn distribution in the bronze-route geometry. In the same set of measurements, a bulk, binary Nb_3Sn needle with a wide range of chemical homogeneity is shown to have almost identical H(T) properties to the commercial ITER wires. This result is surprising since the ternary additions (Ti or Ta) in the commercial wires are expected to enhance Hc2 by at least one Tesla. We compare the performance of this inhomogeneous bulk needle to chemically homogeneous binary and ternary bulk samples.

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Seismic Analysis of The National Compact Stellarator Experiment (NCSX)

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Seismic analysis and qualification of NCSX is presented. DOE requirements as outlined in DOE-STD-1020-2002 are followed for determination of the necessity for seismic qualification of the stellarator and its related systems. IBC-2000 is followed for the qualification requirements. The stellarator presents minimal occupational hazards and hazards to the public. The qualification effort is intended to preserve the viability of continuing the experiment after an earthquake, and to explore the sensitivity of the design to dynamic loading from sources other than normal operation. A response spectra modal analysis has been employed. The model is an assemblage of the simpler models of the vessel, and modular coil shells; being employed to qualify these components for normal operational loading. Outer TF and PF coil models and models of the cold mass supports have been generated and added to form a complete model of the stellarator system. The scale of the model is limited by the computational capacity of the windows/Intel system used for the analysis, and the efforts to control runtimes and file sizes are described. Much of the stellarator is robust to resist normal Lorentz forces. Areas sensitive to lateral loads and dynamic application of non-Lorentz loading, include the nested cylinder cold mass support columns, cantilevered vessel ducts, and the radial guides connecting the vessel ducts and modular coil shell. Loads on these structures are quantified, and design adequacy is assessed. .

Non-linear Analysis of the Modular Coil Windings for the National Compact Stellarator Experiment

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A non linear FEA study has been performed on the modular coils of the National Compact Stellarator Experiment (NCSX). The modular coils provide the primary magnetic field within NCSX and consist of flexible cable conductor wound on a cast and machined winding form and vacuum impregnated with epoxy. Eighteen coils and associated winding forms are connected at assembly into a toroidal shell structure. The purpose of this study was to evaluate the structural response of the windings and shell structure during cooldown and normal operation. Two separate software packages were used for the study, and two independent analyses were undertaken. The first analysis performed with Pro/Mechanica®, examined both the response of the modular coils to magnetic pressure and thermal effects during a 2 Tesla pulse. Modeled items included a portion of the shell structure the winding packs, and a set of 48 "pseudo clamps". The so called "pseudo clamps" are represented simply by blocks of material that are restrained in their respective normal directions and have properties which mimic the stiffness of the spring washers of the actual preload clamps. The winding pack is free to slide on the shell structure and is restrained only by the clamps. A second model, including the complete shell structure of each coil, was studied with the FEA program ANSYS®. Contact regions defined in both Pro/Mechanica and Ansys allow the winding to slide and detach form the shell structure. The two analyses are compared for parameters such as winding/structure gap, overall displacement, equivalent stress and principle strain values.

Electromagnetic Linear Structural Analysis of the National Compact Stellarator Experiment (NCSX) Modular Coil System

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A detailed electromagnetic-structural ANSYS analysis of the NCSX Modular Coil (MC) system is presented. The simplified (linear) model is used to provide some insights into the essential behavior of the modular coil. In the actual device, the winding packs are Vacuum Pressure Impregnated (VPI'd) in-place and restrained by 50+ clamps per coil. In general, JxB Lorentz forces press the WP onto the structure which makes the linear ("glued") approach justifiable. The benefit, of course, is relatively fast computer run-times and a modeling tool which is able to perform numerous design studies. However, there are regions where the EM forces point away from the structure and locally invalidate the "glued" approximation.

The results of a variety of design studies are presented, such as the structural stiffness and worst case running loads at the poloidal breaks, non-ideal coil center displacements from thermal contractions and structural loads, smeared winding pack and winding form stresses, and the effects of supporting the convoluted MC "wings" with the neighboring shell. Critical results are illustrated with contour plots, and where possible, compared to the requirements of the NCSX structural design criteria. The analysis provides the following results:

- Poloidal Breaks are exposed to a tensile running load of 4 k-lb/in to 9 k-lb/in from EM loads.
- The stiffness of the MC shells to opening displacements at the poloidal breaks is 22 k-lb/in to 57 k-lb/in, depending on the shell type ("A", "B" or "C").
- The maximum principal strain in the conductor is determined to be ~0.1%. With the WP in weak compression, or a near zero stress state at the end of the pulse, a zero to 0.1% and back to zero cyclic strain test for 130,000 cycles represents the worst in-service loading.
- The maximum stress in the stainless steel winding forms is determined to be ~190 MPa. Although a design-basis fatigue curve for the SS casting material is work in progress, the stress range is ~1/3rd the yield stress of annealed 316LN, which would indicate near endurance limit behavior.
- Cool-down displacements and MC WF deformations from EM loads are calculated and lead to non-ideal coil positions, off by as much as 1.5 mm. Displacements at each element centroid are provided as input to field error calculations by others.
- The effects of MC module Type C-C mechanical continuity in the inaccessible inboard region at final assembly are studied and show that only toroidal continuity provides any benefit. In-plane restraints (i.e., shear keys) provide essentially no benefit, and therefore eliminate the need for such details.
- Providing support at the extremes of the MCWF "wings" is critical to minimizing the bending stresses in the WP.
- The wing supports must be capable of carrying about 0.6 MN (135 k-lb) in compression with minimal deflection in order to be effective.

Testing of NCSX Composite Coil Material Properties

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The National Compact Stellarator Experiment (NCSX) is now in design and requires 18 modular coils that are constructed to a highly complex geometry. The modular coil conductors are designed as a composite of a fine gauge stranded copper cable shaped to the required geometry and vacuum impregnated with a resin. These composite conductors exhibit unique material properties that must be determined and verified through testing. The conductor material properties are necessary for design modeling and performance validation. This paper will present the methods used to test and measure the coil conductor material properties, the unique challenges in measuring these complex materials at both room and liquid nitrogen temperatures and the results of those tests.

STRUCTURAL ANALYSIS OF THE NEW JET TAE ANTENNA

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In this paper the mechanical design of the new active MHD antennas for JET is described and the structural/mechanical analysis for the antennas is presented. These new antennas replace the existing n = 1 or 2 saddle coils with a set of eight smaller antennas designed to excite Toroidal Alfven Eigenmodes (TAE's) with high toroidal mode number (n ~ 10) in the frequency range of 30 kHz - 500 kHz. TAE's with these higher mode numbers are expected in ITER and could enhance the loss of fast alpha particles in a burning plasma regime. By studying the properties of stable TAE's excited actively by these antennas, high performance regimes of operation avoiding unstable fast particle driven modes can be found. Design details have evolved. Currently one antenna assembly consists of four rectangular windings of 4 mm Inconel 718 wire, with 18 turns, wound on alumina-oxide spool pieces. Two of these assemblies will be installed at toroidally opposite positions. Antenna wires are protected from the plasma heat flux by CFC tiles mounted on mini-limiters, located between the individual windings. The main structural element is a 60 X 120mm Inconel 625 box section. The support scheme utilizes cantilevered brackets that connect to the saddle coils, and “wing” brackets which add support to the top of the frame. Conservative estimates of the disruption currents in the MHD antennas and frame were used to calculate loading and resulting stress in the antenna structure. Fields, field transients, and halo current specifications were provided by JET. The frame originally was designed as a continuous loop, and was converted to an open structure to break eddy current loops. Antenna eddy currents were computed assuming the antenna is shorted. In the latest design, frame forces primarily result from halo currents entering around the mini limiters that now protect the antenna windings. Accelerations due to the vessel disruption dynamic response were included in the loading. The dynamic response of the antenna and frame is computed using a time transient analysis with the loading assumed as half a sine wave.

Wire Debris Modeling of the Z-Accelerator

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The Z-Accelerator is an intense x-ray source located at Sandia National Laboratory. On a typical shot, 20 MA of current passes through a cylindrical array of wires over tens of nanoseconds. The result is the release of 2 MJ of low-energy x-rays at approximately 200 TW. The wires are mostly vaporized in this time, but some wire fragments remain[1]. We have developed a model for the deformation of these wires as they accelerate towards the center of the device. While the shot is generally over 200 nanoseconds, the model only covers times on the order of 1-4 nanoseconds, as it is a continuum model.

The model begins with a 2-D model in a commercial finite element code (ANSYS) that determines the forces and magnetic fields the titanium wires experience early in a typical shot. The magnetic field around the wires reaches a maximum of 210 Tesla when the current is a maximum. ANSYS provides a force per unit length that is applied to the wire over time.

The forces that are determined in ANSYS are used in a separate computer code that solves the equations of motion for the wires. The code solves the 1-D wave equation with a periodic forcing function, using only the early portions of a cycle to approximate a monotonically increasing load. As the wire is displaced from its initial position, the tension should increase as the length of the wire increases. An incremental model is used to update the tension as the wire is displaced, effectively linearizing an inherently nonlinear problem[2].

Results will be described that show the wires' behavior as a function of the initial tension applied to the wire. Also shown will be the estimates of the different fragment sizes remaining after the wire has undergone fragmentation. The fragment sizes will be estimated using existing fragmentation models.

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Structural Analysis of the NCSX Vacuum Vessel

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The NCSX vacuum vessel has a rather unique shape being very closely coupled topologically to the three-fold stellarator symmetry of the plasma it contains. This shape does not permit the use of the common forms of pressure vessel analysis and necessitates the reliance on finite element analysis. The current paper describes the NCSX vacuum vessel stress analysis including external pressure, thermal, and electro-magnetic loading from internal plasma disruptions and bake-out temperatures of up to 400 degrees centigrade. Buckling and dynamic loading conditions are also considered.

DESIGN REQUIREMENT, QUALIFICATION TESTS AND INTEGRATION OF A THIN SOLID LUBRICANT FILM OF MOS2 FOR COLD MASS SUPPORT STRUCTURE OF THE STEADY STATE SUPERCONDUCTING TOKAMAK SST-1.

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The SST-1 is a super conducting tokamak, which is in the final phase of assembly and commissioning. The super conducting magnet system of SST1 comprises of Toroidal field (TF) and Poloidal field (PF) coils. The 16 TF coils are nosed and clamped towards the in-board side and are supported toroidally with inter-coil structure at the out-board side, forming a rigid body system. The 9 PF coils are clamped on the TF coils structure. The integrated system of TF coils & PF coils forms the cold mass of @ 50 Ton weight. This cold mass is accommodated inside the cryostat and freely supported on the rigid support ring at 16 locations and support ring in-turn supported on 8 columns of machine support structure. During the operation this cold mass attains a cryogenic temperature of 4.2K in the hostile environment of high vacuum. The thermal excursion of cold mass and its supporting structure during this cool down results into severe frictional forces at the supporting surfaces. There is a design requirement of introducing a thin layer of solid lubricant film of MOS2 having coefficient of friction 0.05 between the sliding surfaces to control the stress contribution due to the friction.

To ascertain the compatibility of molybdenum disulphide (MOS2) as a solid lubricant in high vacuum and very low temperature environment, we have carried out qualification tests on various samples and measured the coefficient of friction in both the room temperature conditions and at high vacuum & after thermal shocking to 4.2K temperatures. After successful qualification tests actual components are fabricated and integrated in the cold mass support structure assembly. This paper presents the design requirement, qualification tests performed and details about the integration of thin solid lubricant film of MOS2.

Oral Session I-4

Power Plant Studies

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ARIES-IFE Assessment of Operational Windows for IFE Power Plants

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The ARIES-IFE study, a national US effort involving universities, national laboratories and industry was an integrated study of IFE chambers and chamber interfaces with the driver and target systems. Rather than focusing on a single design point, the study aimed at identifying design windows, trade-offs, and key physics and technology uncertainties for various IFE chamber concepts.

An essential of element of such a study is the detailed characterization of the target yield and spectrum. We have selected heavy-ion indirect target designs of LLNL/LBL and direct-drive target designs of NRL as our reference targets. Detailed spectra from these two targets have been calculated -- their photon and ions/debris spectra are vastly different. Our analysis indicates that this detailed information of the target yield and spectrum plays a crucial role in defining the design windows for IFE systems.

Three main classes of chamber concepts are analyzed including dry walls, wetted-walls (thin-liquid-protected) and thick-liquid wall concepts. The three classes of chamber concepts use different schemes to ensure survival of the first wall: gas protection for dry walls or liquid protection for the other two. In each case, survival of the first wall leads to sever constraints on the chamber size and geometry, material choices, and maintenance of chamber protection scheme (*e.g.,* replenishment of liquid protective layer). As a result of interaction of target particle and energy flux with the first wall, material is evaporated or ejected into the chamber. These materials evolve, cool, and are pumped out during the transition between driver shots. The chamber environment prior to the next shot will depend on the evolution of the chamber constituents during the time between shots. A cryogenic target has to be injected and tracked in this chamber and the driver beams should propagate and be focused in this pre-shot chamber environment. Constraints imposed by target survival during the injection process and driver beam transport and focusing can be translated back into requirement on the chamber itself, further restriction the design space available for the chamber.

By overlaying the constraints from various systems such as target injection and tracking, thermal response of the first wall, and laser or heavy-ion propagation and focusing, one can arrive at operational windows for IFE power plants. These constraint and the resultant operational windows are discussed in this paper. This process has allowed us to identify key uncertainties and directions for further R&D.

European Fusion Power Plant Studies

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From 1990 to 2000 studies within the European fusion program examined the safety, environmental and economic potential of fusion power. Since the establishment of the main features of the conceptual designs upon which those studies were based there have been substantial advances in the understanding of fusion plasma physics and plasma operating regimes in tokamaks, and progress in the development of materials and technology. Moreover, the safety and environmental studies were based on designs that were not economically optimized, and the various studies were not intended to be fully consistent with one another in detail. Accordingly, a more comprehensive and integrated study, updated in the light of our current know-how and understanding, was launched, to serve as a better guide for the future evolution of the fusion development program. This was the Power Plant Conceptual Study (PPCS), which reported in the summer of 2004.

Several conceptual designs (“Models”) for commercial fusion power plants were developed, spanning a range from relatively near term to more substantial extrapolations. All are based on the mainstream tokamak concept. All have an electrical output of 1.5 GWe. However, because of differences in their plasma physics and materials basis, they differ substantially in their fusion power, size and re-circulating power fraction. The parameters of the Models were chosen by systems analysis to be economically optimal, given the specifically assigned constraints on plasma and technology performance. Those analyses also show which plasma, materials and engineering parameters are keys to further improving the economics. The conceptual designs were developed in some detail, including innovative aspects: a helium-cooled divertor able to withstand 10 MW/m2 steady heat load and a maintenance concept resulting in 75% availability. Key features will be described in this paper. Analyses were made of the safety, environmental impacts and economic performance of the Models.

The calculated cost of generating electricity from the Models is in the range of published estimates for the future costs from other sources. Even the near-term Models are economically viable. Drawing on more extensive work in the European Socio-Economic Research in Fusion program, the “external” costs were also calculated. External costs are those associated with any environmental damage or adverse effects upon health. These costs were very low, for all the Models: similar to wind power and much less than fossil fuels. Economic optimization of the designs did not jeopardize their safety and environmental performance. All the Models proved to have the attractive and substantial safety and environmental advantages shown in earlier studies, now established with greater confidence.

The results for the near-term Models suggest that a first generation of fusion power plants – those that would be accessible by a fast track route of fusion development, going through ITER and not entailing major materials advances – will be economically acceptable with major safety and environmental advantages. The remaining results illustrate the potential for more advanced power plants. It is concluded from the PPCS results that the main thrusts of the European fusion development program are on the right lines.

Synergies between Generation-IV and Advanced Fusion Power Plants

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For the first time since the early 1990's, the U.S. Department of Energy has long term research and development programs in both nuclear fission and nuclear fusion, the Generation IV research and development program and the ARIES program, respectively. Moreover, they are both working towards energy sources with ambitious sets of constraints and a great deal of promise. The synergies created by these increasingly parallel programs open the door for new collaborations that will increase the total effectiveness of research needed in both. This paper reviews some of the most important opportunities.

Materials

The most obvious, and arguably the most important, area of collaboration is that of materials research and development. While advanced fusion power plant studies have, for some time, required low-activation materials that can operate at extreme temperatures (700-1000^{o}C) in a high energy radiation environment, most fission power plants developments have been content to limit their environments to the lower temperatures and thermal fluxes of a light water reactor environment. The Generation IV program changes that with an emphasis on very high temperature cycles combined, in some cases, with faster spectra. Innovations in materials research such as computational materials design and novel materials testing capability will benefit both communities as they attempt to achieve longer component lifetimes and more benign waste streams.

Safety and Regulation

The ARIES fusion power plant studies have long included a constraint that leads to an enhanced safety philosophy: no off-site emergency response to any design basis accident. The Generation IV program adds this constraint to the development of future fission reactor systems. At the same time, the regulatory framework, currently focused on the safe operation of light water reactors, will require changes to accommodate the Generation IV fission reactors and ultimately the fusion power plants. Similarly, new approaches to the classification of waste streams to minimize the amount of material destined for geologic disposal will assist both communities in meeting their waste management goals. The process of enacting those changes, as well as the specific changes in some cases, both represent an opportunity for mutual benefit.

Economics and Energy Products

It is difficult to predict the energy markets of the future, but most fusion and fission systems can be expected to be deployed at large power levels and require large capital investment in order to facilitate a competitive cost of electricity. As future energy markets develop, some of the economic analysis for advanced nuclear systems will be common to all systems. This is particularly true when considering the role of nuclear systems in alternative/emerging energy markets such as the production of hydrogen fuels.

FIRE, A Test Bed for ARIES-RS/AT Advanced Physics and Plasma Technology

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The overall vision for FIRE is to develop and test the fusion plasma physics and plasma technologies needed to realize capabilities of the ARIES-RS/AT power plant designs. The mission of FIRE is to attain, explore, understand and optimize a fusion dominated plasma which would be satisfied by producing DT fusion plasmas with nominal fusion gains ~10, self-driven currents of ≈ 80%, fusion power ~ 150 - 300 MW and pulse lengths up to 40 s. Achieving this goal will require the deployment of several key fusion technologies under conditions approaching those of ARIES-RS/AT.

The FIRE plasma configuration with strong plasma shaping, a double null pumped divertor and all metal plasma facing components is a 40% scale model of the ARIES-RS/AT plasma configuration. "Steady-state" advanced tokamak modes in FIRE with high β ($\beta_N \approx 4$), high bootstrap fraction ($f_{bs} \approx 80\%$) and 100% non-inductive current drive are suitable for testing the physics of the ARIES-RS/AT operating modes. The FIRE AT mode utilizes Fast Wave Current Drive (FWCD) for on-axis current drive and Lower Hybrid Current Drive (LHCD) for off-axis current drive with no external momentum input, similar to the systems envisioned for ARIES. Although inductive and non-inductive current drive are used to ramp the plasma current, the flattop plasma has a "steady-state" 100% non-inductive current provided by the combination of bootstrap, lower hybrid, and fast wave current, and the current profile is held constant for 3.2 τ_{CR}. Both FIRE and ARIES-RS/AT would rely on passive stabilization from close fitting conducting structures and Resistive Wall Mode (RWM) stabilization from coils mounted just behind the first wall structure. The development of closely coupled RWM coils for an environment with neutron fluxes similar to ARIES-RS/AT will be an important contribution of the FIRE program.

The removal of plasma exhaust power is a major challenge for a magnetic fusion power plant, and the development of techniques to handle power plant relevant exhaust power is a major objective for a burning plasma experiment. The FIRE-AT $\beta \approx 4\%$ would result in fusion power densities in FIRE from 3 - 10 MWm^{-3} and neutron wall loading from 2 - 4 $MW\ m^{-2}$ which are at the levels expected from the ARIES–RS /AT design studies. The divertor and first wall thermal loads would also be in the range expected for ARIES. The FIRE tungsten divertor has the capability of handling steady-state thermal loads approaching those of ARIES while maintaining low tritium inventory. The first wall design of FIRE features Be tiles that are capable of absorbing 1 MWm^{-2} for ~40s, and would provide a good test of whether Be would be suitable as a first wall material in ARIES-RS/AT. The effective tritium retention in the plasma facing components (PFC) of ARIES must be < 0.04% to allow operating periods of ≈ one year before an intervention to remove tritium. This is much less than the typical 10-30% tritium retention observed in the DT experiments on TFTR and JET, which had carbon PFCs. FIRE would be able to provide a good test of the feasibility of a W/Be divertor first wall design for ARIES-RS/AT. FIRE utilizing articulated boom remote manipulators would provide remote handling experience of internal plasma facing components.

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Progress Toward Development of an IFE Power Plant Using Z-Pinch Technology*

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The Z-Pinch Power Plant uses the results from Sandia National Laboratories' Z accelerator in a power plant application to generate energy pulses using inertial confinement fusion. A collaborative project has been initiated by Sandia to investigate the scientific principles of a power generation system using this technology. Research is underway to develop an integrated concept that describes the operational issues of a 1000 MW electrical power plant. Issues under consideration include: 1-20 gigajoule fusion pulse containment, repetitive mechanical connection of heavy hardware, generation of terawatt pulses every 10 seconds, recycling of ten thousand tons of steel, and manufacturing of millions of hohlraums and capsules per year. Additionally, waste generation and disposal issues are being examined. This paper will describe the current concept for the plant and describe the objectives for future research.

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Helical Fusion Power Plant Economics Studies

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The Physics-Engineering-Cost (PEC) Code has been updated to include data from specific blanket-shield designs, a new costing structure, more recent unit costs, and improved algorithms for physics and engineering parameters. For benchmarking the Code has been used to model the ARIES-AT tokamak and the ARIES Stellarator Power Plant Study (SPPS). The Cost of Electricity (COE) variations with various parameters are studied.

The COE is related strongly to the neutron wall load, which can be enhanced by increasing the magnetic field, the output power, or the value of beta. At a given output power level, increasing the value of beta reduces the reactor size, which reduces the energy confinement time, making ignition more difficult. Therefore, the useful value of beta is limited by the ignition condition and the desired output power, in addition to the beta limits from plasma equlibrium and stability.

There is a strong economy of scale, showing that Heliotron power plants can be economicaly competitive at high power levels, so the siting issues of large power plants are important.

Oral Session I-5

ITER Test Blanket Modules - Special Session

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Interface of Blanket Testing and ITER design.

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One of the objectives of ITER is to demonstrate fusion technology in an integrated system by performing testing of nuclear components required to utilize fusion energy, in particular- to test design concepts of tritium breeding blanket relevant to a ` DEMO reactor. In the current ITER design three equatorial ports which can accommodate modules with the face cross section up to 1310 mm wide x 1760mm high have been allocated for blanket modules testing.

Typical testing conditions foreseen now include a surface heat flux of 0.25 MW/m^2, a neutron wall load of 0.78 MW/m^2, a pulse length of 400 s with a duty cycle of 25%. After first 10 years of operation one may expect to reach the total neutron fluence at the surface of test blanket modules (TBMs) ~0.12 Mwy/m^2.

Progress in physics of hybrid and non-inductive current drive may lead in the second 10 years of operation to very long pulses (many thousands seconds) with a similar neutron wall loading and accumulation of neutron fluence at least up to 0.3 MWy/m2.

TBMs will be mounted on water–cooled steel frames with additional shielding and inserted in equatorial ports by standard ITER remote handling equipment for installation , removal and transportation of port plugs to and from a hot sell.

The tritium measuring equipment will be located in containers in front of ports. The heat exchangers and safety (the pressure suppression tank for the water-cooled TBM) equipment may be placed inside of the ITER Tokamak Cooling Water System vault if the space in front of ports is not enough . Tritium removal equipment may be installed in Tritium building.
In special cases some equipment may be placed inside the ports (for example liquid lithium loop to minimize amount of lithium. Heat rejection and accumulation of produced tritium will be provided by ITER.

TBMs replacements will be synchronized with ITER operation and will be performed in the ITER hot cell, where the whole TBMs/shield plug systems will be remotely transported. Arrangements for Hot sell post irradiation examination will depend on selected site or further decisions . Currently the ITER hot sell is not big enough and not properly equipped for such a task .

Test modules must not compromise ITER safety and reliability . As a result they must satisfy several requirements depending on their design. Water-cooled modules must not leak frequently inside the main vacuum vessel and must have their own pressure suppression system.
Helium leaks are permitted. Mass of liquid lithium is strictly limited to avoid hydrogen explosion .

Pressure to reduce the ITER cost has lead to decreased neutron flux and fluence and to more limited testing conditions. Nevertheless, performed analysis indicates that breeding blanket testing in ITER is extremely important for DEMO breeding blanket development .The best effort has to be done to coordinate parties activities in this area and to achieve the best use of space and time available for blanket testing in ITER.

US Plans and Strategy for ITER Blanket Testing

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A critical element in the ITER mission since its inception has been testing integrated blanket modules in special ports. Among the principal objectives of the ITER Test Blanket Module (ITER-TBM) Program are: 1) develop the technology necessary to install breeding capabilities to supply ITER with the tritium necessary for operation in its extended phase of operation; and 2) provide experimental data vital to evaluating the feasibility, constraints, and potential of the D-T cycle for fusion systems (including limitations on options for improving plasma physics performance, e.g., conducting shells, passive coils, thick armors/first wall). Adequate tritium supply is a central issue for the operation of ITER and the development of fusion energy. TBMs will be inserted in ITER from Day 1 of its operation and will provide the first experimental data on the feasibility of the D-T cycle for fusion.

With the US rejoining ITER, the US community has decided to have strong participation in ITER-TBM. The US has been a leader in the science and engineering of technology testing on ITER and other fusion devices and has many unique capabilities to contribute to the ITER-TBM program. A US strategy for ITER-TBM has evolved that emphasizes international collaboration. A study was initiated to select the two blanket options for the US ITER-TBM in light of new R&D results from the US and world programs over the past decade. The study is led by the Plasma Chamber community in partnership with the Materials, PFC, Safety, and physics communities. The study focuses on assessment of the critical feasibility issues for candidate blanket concepts and it is strongly coupled to R&D of modeling and experiments. Examples of issues are MHD insulators, SiC insert viability and compatibility with LiPb, tritium permeation, MHD effects on heat transfer, solid breeder "temperature window" and thermomechanics, and chemistry control of molten salts.

The initial conclusion of the US community, based on the results of the technical assessment to date, is to select two blanket concepts for the US ITER-TBM with the following emphases:

1. Select a helium-cooled solid breeder concept with ferritic steel structure and neutron multiplier, but without a fully independent TBM. Rather, plan on unit cell and submodule test articles that focus on particular technical issues of interest to all parties. (All ITER Parties have this concept as one of their favored options.)
2. Focus on testing Dual-Coolant liquid breeder blanket concepts with ultimate potential for self-cooling. Develop and design TBM with flexibility to test two options:
    a. a helium-cooled ferritic structure with self-cooled LiPb breeder zone that uses SiC insert as MHD and thermal insulator (insulator requirements in dual-coolant concepts are less demanding than those for self-cooled concepts);
    b. a helium-cooled ferritic structure with low melting-point molten salt. The choice of the specific lithium-containing molten salt will be made based on near-term R&D experiments and modeling. Because of the low thermal and electrical conductivity of molten salts, no insulators are needed.

European Test strategy for Test blanket modules to be tested in ITER

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There are two European Test Blanket Modules (TBM) a Helium Cooled Pebble Bed (HCPB) and a Helium Cooled Liquid Lead (HCLL). During 2003 both concepts were redesigned with the goal to use as much as possible similar design options and fabrication techniques for both types in order to reduce the European effort for TBM development. The result is a robust TBM box being able to withstand 8 MPa internal pressure in case of in-box LOCA, filled with typically 18 and 24 breeding units (BU), for HCPB and HCLL respectively. A breeding unit has about 200 mm in poloidal and toroidal direction and about 400 mm in radial direction. The TBM box consists of First wall (FW), caps, stiffening grid and manifolds. The BU's have a back plate and different types of cooling plates in the radial toroidal plane.

The test strategy starts with the qualification of the fabrication processes for the important subcomponents. These are Hot Isostatic Pressing (HIP) to make the caps, stiffening grid and FW, for the latter also the bending of the plate has to be tested. Another important item is the qualification of TIG welding process with additive wire. This is new for the EUROFER steel to be used for the TBM's. Clarification of this question has an important feedback on the weld design and preparation. Finally a small demonstrator is to be produced where FW, stiffening grid and caps are welded together to see the quality of the welds produced under more realistic conditions.

In the next step special problems were investigated like a) flow distribution in the TBM manifolds and the mass low distribution in all the channels; b) thermal stresses of pebble beds in a BU interacting with the adjacent cooling structure c) losses for the HCLL flow in a magnet filed under typical conditions.

The third type of tests are integral tests of testing in large Helium facilities simulating the boundary conditions in ITER. These steps are mandatory to qualify such a component which should be inserted in ITER without jeopardising availability or safety of this machine.

For the tests in ITER of the HCPB concept four different types are foreseen. The first, the so called Electro Magnetic (EM) module is used in the initial phase of ITER, without neutron flux. This allows checking the computer codes, used for predicting the electromagnetic forces. At the same time typical characteristics of the Helium coolant system can be tested.

The second module, used from the D-D plasma phase of ITER is to test the codes for the neutron transport and interaction with the module and the Tritium generation rate.

The third module is to check the knowledge of pebble bed behaviour and its interaction with the steel cooling plates. Thermal properties and mechanical questions have to be answered, for the latter especially the interplay of swelling and creep under the constraints of temperature field and mechanical enclosure.

The fourth module finally is thought to be kind of a demonstrator. It should show the possibility of breeding sufficient Tritium for continuous power process, the high grade of heat being extracted and the stable operation under different loading conditions.

Plan and strategy for ITER Blanket Testing in Japan

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This paper presents a plan and strategy for ITER blanket module testing in Japan. The Fusion Council of Japan has established the long-term research and development program of the blanket in 1999. In the program, the solid breeder blanket was selected as the primary candidate blanket of the fusion power demonstration plant in Japan. In the program, Japan Atomic Energy Research Institute (JAERI) has been nominated as a leading institute of the development of solid breeder blankets, in collaboration with universities, for the near term power demonstration plant, while, universities including National Institute for Fusion Science (NIFS) are assigned mainly to develop advanced blankets for longer term power plant development.

In the long term research and development program, ITER blanket module testing is identified as the most important milestone, by which integrity of candidate blanket concepts and structures are evaluated. For establishing material irradiation data for fusion power plants, International Fusion Material Irradiation Facility (IFMIF) is expected to provide the mechanical data of Reduced Activation Ferritic/martensitic Steel, silicon carbide composite, vanadium alloy, ODS ferritic steel and other functional materials, for material selection and licensing data base. In Japan, universities, NIFS and JAERI cover a variety of types of blanket development. In the Test Blanket Working Group, Japan showed interest to all working sub-groups.

As the primary blanket options, solid breeder test blankets with ferritic steel structure cooled by helium and water are being developed by JAERI with cooperation of university experts. Japan has indention to fabricate test modules for solid breeder blankets in a possible collaboration with other international partners as appropriate. In all necessary fields of blanket development, element technology development phase has been almost completed and is now stepping further to the engineering test phase, in which scalable mockups of solid breeder test blanket modules will be fabricated and tested to clarify the total structure integrity for final specification decision of test blanket modules.

As the advanced blanket options, solid breeder blanket module with SiC composite structure cooled by high temperature helium gas, liquid LiPb breeder cooled by helium, molten salt self cooled blanket module and liquid Li self cooled blanket module are under development by universities and NIFS with cooperation of JAERI. Key issues have been addressed and critical technologies are being developed.

The development of blankets in Japan has shown sound progress on both of solid and liquid breeder blankets under coordinated domestic development programs, for both of primary and advanced options.

Thermofluid Magnetohydrodynamic Issues for Liquid Breeders

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Breeding blanket designs utilizing liquid breeder materials have been considered for fusion for many years. But since 1992, the only self-cooled blanket concept envisaged for tests in ITER was the lithium (Li) breeder/coolant, vanadium structure, coated insulator barrier concept. The situation has changed recently by the inclusion of self-cooled Molten Salt (MS) concepts and various “Dual Coolant” concepts with lead-lithium (PbLi) or MS as breeder/coolant. These new concepts have new magnetohydrodynamic-related (MHD) issues to be tested in ITER.

This paper provides a description of the most promising liquid breeder blankets currently proposed for testing in ITER, including:

(a) Self-cooled Lithium with vandium structure and insulator barrier,
(b) Separately-cooled PbLi with ferritic steel structure,
(c) Dual coolant PbLi with ferritic steel structure and SiC flow channel inserts, and
(d) Various incarnations of separately-cooled and dual coolant MS with ferritic steel structure.

The critical MHD issues for self-cooled and dual coolant liquid metal systems are the MHD pressure drop and flow distribution with ideal and imperfect insulator barriers/coatings, ideal and imperfect free-floating flow channel inserts, and complex geometry flow elements like expansions, contraction, manifolds, *etc*. Separately-cooled PbLi systems still must circulate the PbLi for tritium removal, and similar MHD issues may limit the flow velocity and influence the tritium permeation due to creation of stagnant regions and other non-ideal flow distribution effects. Molten salt breeder/coolants have significantly reduced electrical conductivity as compared to liquid metals and MHD pressure drop is not considered a serious issue. However, MS also have much lower thermal conductivity as well, and the heat transfer to/from the structure depends on turbulent convection. The degradation of convective heat transfer by MHD turbulence modification/suppression is of great interest for both self-cooled MS systems where first wall cooling may need to be enhanced, and dual coolant MS systems where heat transfer from the hot breeder to the cooler wall needs to be suppressed.

An assessment is provided as to which of these MHD-related issues of liquid breeder blankets can be investigated in ITER, along with a discussion of the strong points and limitations of MHD-related tests in ITER. A coordinated plan is evolved for testing needed prior to ITER, tests conducted during the first ITER phase with no neutrons, and later phase tests coupled to heat transfer effects and the thermomechanical loading of structures with electroinsulating barriers/coatings and inserts.

Engineering Scaling Requirements for Solid Breeder Blanket Testing

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An engineering scaling process is proposed for ITER TBM and applied to the various designs in accordance with the testing objectives of validating the design tools and the database, and evaluating blanket performance under Demo operational conditions. The goal is to improve design of solid breeder blanket test modules based on scaling law practices. The focus of this study is to address key issues concerning pebble bed thermomechanical and first wall structural response, as well as neutronic performance. The goal of scaling is to ensure that changes in structural response and performance caused by changes in size and operating conditions does not reduce the usefulness of the tests. In addition, in order to evaluate statistical significance, the question of to what extent the size of the test article can be reduced such that a multiple number of test articles can be tested simultaneously is addressed. Since there is no simple analytical scheme that can be applied, brute-force computational approaches, including FEM, are utilized to recapture key Demo parameters under a reduced neutron wall load in ITER.

Reproducing Demo operating temperature is key to ITER test module design since it has a crucial influence on solid breeder blanket performance. However, complexities arise, since the issue is driven not only by temperature magnitudes but also by temperature gradients, which affect pebble bed performance under thermomechanical loads. For example, temperature gradients determine the stress magnitude exerted on the coolant wall, yet temperature magnitudes influence the rate of bed creep strain, which results in reducing stress magnitude once bed compaction initiates. Initial analysis of a design in which the Demo-like temperature magnitude is preserved shows an increase in stress magnitude of 15%. The pulsed operation nature of ITER further complicates the engineering scaling design process, due to the alteration of the thermal time constant. Ultimately, the size of the test module is greatly impacted by the trade-off between these competing factors. Neutronics performance scaling issues aim to achieve high confidence on tritium production rate prediction (i.e. ±5% relative to the characteristics in the full coverage case found in Demo). Geometrical requirements dictate scaling down the test module to a small size to ensure high spatial resolution for a specific measurement, taking into account the presence of the test port frames and divider plates. Preliminary results show that a test module of ~45 cm in the poloidal direction and ~60 cm in the toroidal direction may achieve the neutronic accuracy goals. As for the first wall performance, the issue is whether Demo characteristic stress and deformation magnitudes can be preserved by modifying geometric parameters. To ensure Demo relevant testing conditions a FEM analysis is employed to determine necessary changes to the ITER TBM design.

Oral Session I-6

Non-Electric Applications

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Fusion Production of Hydrogen; How Fusion Energy Can Fuel the *Hydrogen Economy*

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The United States has embarked on a serious effort to transform our transportation economy from one that is largely petroleum-based to one based on hydrogen. This has come to be known as *the Hydrogen Economy*. If successful, this transition will result in significant improvements in energy efficiency and environmental quality. A hydrogen economy can be based on domestic energy resources and would make possible a high degree of energy security.

Hydrogen is an energy carrier, not an energy source. While hydrogen is the most plentiful element on earth, virtually all of it is chemically bound. Energy must be invested to separate hydrogen from the water, hydrocarbons or carbohydrates in which it is bound. The most straightforward, cleanest and sustainable pathway to hydrogen is decomposition of water. This can be accomplished by electrolysis using electricity, by high temperature electrolysis using both heat and electricity, and by a variety of thermochemical water-splitting cycle processes using only heat. Radiolysis is a potential technique for splitting of water that could use fusion energy directly to make hydrogen.

Fusion energy could be the ultimate best source of the energy needed to make the vast amounts of hydrogen needed for a hydrogen economy. Several studies done over the years have all concluded that production of hydrogen is well suited to the characteristics of fusion energy production, and could be a larger market for fusion energy than even electricity production. These studies have shown that electrolysis, high temperature electrolysis and thermochemical water-splitting all have the potential to be attractive techniques for the production of hydrogen using fusion energy.

The DOE hydrogen program is currently developing these techniques, and is also developing high temperature nuclear fission reactors that could use these techniques for hydrogen production. Fusion can take benefit from this development. Use of fusion for low temperature electrolysis will have no impact on the fusion designs envisioned for electricity production. High temperature electrolysis and thermochemical water-splitting, which offer the potential for higher efficiency and lower costs, would have impact on the fusion designs and would add additional requirements and constraints to the already difficult fusion reactor design process. Strict control of tritium to avoid contamination of the hydrogen product will be especially important. Several fusion design concepts have been developed that appear to successfully meet the requirements for hydrogen production.

Production of hydrogen for the *Hydrogen Economy* is an attractive mission for fusion energy and could be a much larger ultimate use of fusion than electricity production. Special fusion reactor designs will be needed for high efficiency production of hydrogen, but low temperature electrolysis could be used with no constraints on fusion design. Fusion does have the potential to provide the ultimate source of fuel for the *Hydrogen Economy*.

Potential Fusion Market for Hydrogen Production Under Environmental Constraints

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As a candidate of future energy, fusion must consider its possible market to be deployed. It is anticipated that the supply of limited reserve of fossil fuel will be tighter while it will not be exhausted in the near future. At the same time, constraint of environmental problem, particularly suspected global warming due to the increase of green house gas such as carbon dioxide will require a shift of energy source from fossil resources to carbon-free technologies. This is the major reasoning for most of the new energy technologies. It should be noted that a large fraction of energy consumption in the future remains in the form of fuel for heat and transportation. Synthetic fuels, particularly hydrogen is regarded as a major secondary energy medium that may have larger market share than electricity. In order to maximize the benefit of fusion, technical possibility of non-electric use of its energy must be evaluated, and it is the objective of the present study.

Hydrogen production processes that can be applied for fusion reactor are compared. Electrolysis of water is one of the straight forward choice, because fusion is primarily considered as source of electricity. However although it is a conservative and established technology, it cannot expect very high efficiency. Unlike light water reactors or renewables, fusion can generate high temperature by selecting blanket design and materials regardless whether it is MCF or ICF. If fusion reactor can generate temperature above 900 degree C, processes such as high temperature vapor electrolysis, thermochemical decomposition of water, steam reforming and other chemical reactions can be considered. Vapor electrolysis, combined with high temperature gas turbine will be efficient and regarded as an adequate technology to be combined with fusion when it is in the phase of commercial use. Chemical reaction of biomass with water vapor; $C_nH_{\square n}O_n...+2nH_2O \rightarrow nCO_2 + 3nH_2$ generates hydrogen from abundant renewable resource, and is proposed as one of an attractive energy use for fusion. Large endothermic energy of 120KJ/mol effectively converts fusion heat to chemical energy. Exhausted carbon dioxide is regarded as carbon neutral, that does not release additional carbon dioxide but recycles it from and to the atmosphere. There will be additional technical issues to be developed if such a process is planned as a possible energy use of fusion. Fusion blanket to be operated at higher temperature is needed, and the chemical process to use eternal heat source must also be developed, with particular attention on tritium contamination.

Socio-economic impacts of fusion is also analyzed and its significance in the global energy market and environmental strategy for the cases of both electricity and hydrogen supply. Unlike in the case of electricity generation, hydrogen market is far more complicated because it is affected by other raw materials, fossil sources, and infrastructure for storage, delivery and use. However it is estimated to increase fusion share in the global primary energy sources to at least 1.5 times larger. In any case, fusion energy must pay attention on the social aspects and continuously consider its possible future market and be flexible to adopt itself to the market.

Tokamak Neutron Source Based Spent Nuclear Fuel Transmutation Reactors

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The (neutron) transmutation of spent nuclear fuel is a potential intermediate-term goal for the magnetic fusion program. We have made design scoping studies for a series of sub-critical fast reactors that would be driven by tokamak fusion neutron sources. These studies were constrained to use existing fusion physics and technology (except for the AT case) and to use reactor and materials technologies that either exist or are being developed in the nuclear program. As shown in Table 1, the ITER physics database and the existing fusion technology are adequate for the design of a fusion neutron source for this application, with the exception of the bootstrap current/current drive efficiency needed for steady-state operation.

Table 1 Tokamak Neutron Source Parameters for Transmutation Reactors

Parameter	FTWR[a]	FTWR-SC[b]	FTWR-AT[c]	GCFTR[d]	ITER[e]
Fusion power, P_{fus} (MW)	≤ 150	≤ 225	≤ 500	≤ 200	410
Neutron source, $S_{fus}(10^{19}$ #/s)	≤ 5.3	≤ 8.0	≤ 17.6	≤ 7.1	14.4
Major radius, R (m)	3.1	4.5	3.9	4.2	6.2
Minor radius, a (m)	0.9	0.9	1.1	1.0	2.0
Elongation, κκ	1.7	1.8	1.7	1.8	1.8
Current, I (MA)	7.0	6.0	8.0	7.2	15.0
Magnetic field, B (T)	6.1	7.5	5.7	6.3	5.3
Confinement, H(y,2)	1.1	1.0	1.5	1.0	1.0
Normalized beta, β_N	≤ 2.5	≤ 2.5	4.0	≤ 2.5	1.8
Plasma Power Mult., Q_p	≤ 2.0	≤ 2.0	4.0	2.9	10
Electric Power Mult, Q_e	1	5		3.6	
Current-drive effic.η_{cd} (A/W)	0.03	0.024	0.05	0.078	
" , γ_{cd} (10^{20} A/Wm^{-2})	0.19	0.20	0.28	0.5	
Bootstrap I fraction, f_{bs}	.67(.38)[f]	0.56(0.24)	0.25	0.35	
Neut. flux, Γ_n (MW/m^2)	≤ 0.8	≤ 1.0	≤ 1.7	≤ 0.9	0.5
Heat flux, q_{fw} (MW/m^2)	≤ 0.4	≤ 0.3	≤ 0.5	≤ 0.3	0.15
Availability (%)	≥ 50	≥ 50	≥ 50	≥ 50	

[a] ITER physics, LN-cool Cu magnets, PbLi cool, metal fuel fast reactor.(FS&T,41,116,2002).
[b] ITER physics, SC magnets, PbLi cooled, metal fuel fast reactor. (FS&T, 45, 55, 2004).
[c] AT physics, SC magnets, PbLi cooled, metal fuel fast reactor. (Ga. Tech rpt GTFR-167, 2003).
[d] ITER physics, SC magnets, He cooled, coated pellet fuel fast reactor (NT, sub., 2004).
[e] ITER design parameters.
[f] required (estimated from present database)

Research and Development of Landmine Detection System by a Compact Fusion Neutron Source

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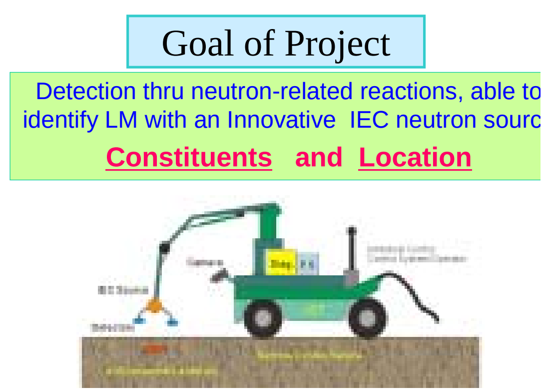
Current results are described on the research and development of the advanced anti-personnel landmine detection system by using a compact discharge-type fusion neutron source called IECF (Inertial-Electrostatic Confinement fusion). It is urgently requested to clean up the lands contaminated by a huge amount of anti-personnel and tank landmines to resume the peaceful citizen lives in more than 60 countries as soon as possible. However, due to the modern landmines made of all plastics tend to hinder quick clearance work by the deminers, since the conventional landmine detectors, like metal detectors, are never effective enough to such plastic mines.

This study started by making use of the technique for BNCT (Boron Neutron Capture Therapy) for cancer treatment in Japan as one of the viable and advanced detection methods of landmines in the international Afghanistan reconstruction program, i.e., detection of backscattered neutrons to identify hydrogen anomaly, and of specific-energy neutron captured γ-rays by hydrogen and nitrogen atoms to identify landmine explosives.

For this purpose, various studies were made, such as a new ion production scheme, i.e., magnetron discharge, for drastic improvement of neutron yields more than 10^8 n/s in pulsed operation including R&D of robust power source, as well as analyses of envisaged detection system with multi-sensors to show promising and practical features for landmine detection in Afghanistan.



A remote-controlled vehicle with a compact fusion neutron source and multi-sensing system

RF Ion Source-Driven IEC Design and Operation

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The IEC (Inertial Electrostatic Confinement) concept has the long range potential for a low-Q fusion reactor. However, in the near term, the has been demonstrated to offer a very attractive compact MeV neutron source in the 10^{8-9} neutron/sec range for various uses such as neutron activation analysis (NAA). Simple changes in operation of the same unit can also provide MeV protons or intense keV X-rays. Such an IEC neutron source has already been used for industrial of various metallic ores during unloading on a conveyer belt. In this case, the IEC replaced a Cf-252 source to provide on-off capability. If the IEC yield is increased by several orders of magnitude, a number of exciting new applications become practical, including use in hospital production of radioisotopes for medical applications. This would also be an important step towards scale-up to reactor level operation.

This paper presents the recent studies aimed at achieving higher yield IECs using a unique external ion source ILLIBS (Illinois Ion Beam Source). ILLIBS employs a RF-driven plasma in an graded magnetic field configuration. Use of this gun allows initiation of the plasma discharge below the normal (Paschen curve) break down region where losses due to charge exchange are greatly reduced. These results suggest that with D-T gas mixture in this unit, a neutron rate of 7.3×10^{12} n/sec at 1.2 mTorr, 75 kV and 1.5 A ion current, should be obtainable with production efficiency 5.9×10^{7} n/J. This represents a significant improvement in neutron production efficiency. Experimental studies of the ILLIBS ion source will be provided along with data for it's operation on the IEC. Implications for future scale-up to higher fusion power levels will be presented.

Overview of University of Wisconsin Inertial-Electrostatic Confinement Fusion Research*

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In Inertial Electrostatic Confinement (IEC) devices, a large voltage difference between concentric, nearly transparent grids in spherical or cylindrical geometry accelerates ions to fusion relevant velocities. Geometric convergence, in theory, leads to a high-density core and increased fusion reaction rates. The University of Wisconsin operates two IEC devices, a cylindrical aluminum chamber and a spherical, water-cooled, stainless-steel chamber, with a power supply capable of 75 mA and 200 kV. The objective of the research program is to generate fusion reaction products efficiently for various applications, including protons for creating radioisotopes for nuclear medicine and neutrons for detecting clandestine materials. One UW IEC device has already produced measurable quantities of proton-produced ^{13}N and ^{94m}Tc, and it has also generated 1.8×10^8 neutrons per second. The UW IEC devices also serve as materials test beds for ions impacting surfaces at 10-200 kV.

Most IEC devices worldwide, including the UW devices, presently operate primarily in a pressure range (1-10 mtorr) that allows ions to make only a few passes through the core before they charge exchange and lose substantial energy or they collide with cathode grid wires. It is believed that fusion rates can be raised by increasing the number of passes and operating at a pressure where ion flow is not impeded by neutral gas. To that end, a helicon ion source has been developed to explore operation at pressures of ~0.05 mtorr, with ion currents up to 30 mA.

The UW IEC research group uses standard proton detectors, neutron detectors, residual gas analyzers, and spectroscopic diagnostics. New diagnostic techniques have also been developed, including eclipse disks to localize proton production and chordwires to estimate ion flux using power balance. Theoretical and computational efforts are underway to provide a better understanding of IEC core physics and fusion reaction rates. These model ion flow dynamics, space-charge buildup, charge exchange, ionization, neutral collisions, and attenuation by the cathode grid.

* Research supported by the Grainger Foundation, the Greatbatch Foundation, and the University of Wisconsin.

Wednesday

September 15, 2004

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Plenary Session II

Fusion Development and Near-Term Facilities

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Historical Perspective on the United States Fusion Program

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Progress and Policy will be traced over the approximately 55 year history of the U. S. Fusion Program. The classified beginnings of the effort in the 1950s ended with declassification in 1958. The effort struggled during the 1960s, but ended on a positive note with the emergence of the tokamak and the promise of laser fusion. The decade of the 1970s was the “Golden Age” of fusion, with large budget increases and the construction of many new facilities, including the Tokamak Fusion Test Reactor (TFTR) and the Shiva laser. The decade ended on a high note with the passage of the Magnetic Fusion Energy Engineering Act of 1980, overwhelming approved by Congress and signed by President Carter. The Act called for a “$ 20 billion, 20-year” effort aimed at construction of a fusion Demonstration Power Plant around the end of the century. The U. S. Magnetic Fusion Energy program has been on a downhill slide since 1980, both in terms of budgets and the construction of new facilities. The Inertial Confinement Fusion program, funded by Department of Energy Defense Programs, has faired considerably better, with the construction of many new facilities, including the National Ignition Facility (NIF).

In 1990, a Fusion Policy Advisory Committee (FPAC) of the DOE Energy Research Advisory Board (ERAB), tried to reset the program back on to a focused track to fusion power. They called for “two distinct and separate approaches, magnetic fusion energy (MFE) and inertial fusion energy (IFE), both aimed at the same goal of fusion energy production.” They said “Both MFE and IFE should increase industrial participation to permit an orderly transition to an energy program with strong emphasis on technology development.”

The Congressional budget cutting frenzy of the mid 1990s will be described, with the nearly disastrous results it had on the MFE program and the curtailment of energy-oriented IFE efforts. The mandated shutdown of TFTR and the U. S. withdrawal from ITER nearly destroyed the logic of the U. S effort. A science-oriented rationale, developed to save the program, had the unanticipated consequence of setting up the magnetic fusion technology and IFE efforts for near elimination. The role of selective Academy reviews in re-enforcing an anti-technology policy climate will be described. A 1999 report by the Secretary of Energy Advisory Board (SEAB), stating “It is our view that we should pursue fusion energy vigorously,” has been largely ignored.

Recent efforts to get the program back on track will be discussed, including the preparation of a 35-year plan, the U. S. rejoining of the ITER international collaboration and the efforts of Congress to ensure that the U S. maintain both fusion technology and IFE efforts.

Time permitting, the history of Fusion Power Associates (FPA) will be described. From its industry-oriented beginning in 1979, FPA has evolved to a laboratory/university orientation as opportunities for industrial participation vanished and the program lost its power plant development focus. FPA remains committed, however, to the view that “Engineering sciences, technology development, systems analysis and plasma sciences should all be considered essential elements in a balanced fusion effort.” [FPA Board of Directors Policy Statement]

The Role of Industry in Fusion Development

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Abstract

The role of industry in US fusion development to date has been substantial, yet with few exceptions in which industry designed and built the entire facilities, the role has been limited to the supply of materials, components and systrems for the various research reactors.

The development of fusion worldwide is proceeding along two main paths: national ones which focus on developing alternate concepts and explore different physics regimes that offer promise for improved operations and less costly commercial reactors, and the other the international effort on ITER and perhaps complementary facilities.

Industry must and can support both paths, but whereas the support of the national programs can continue more or less along the same lines as in the past, there is an opportunity for a substantially different role in ITER.

The role that U.S. industry can play in both the U.S domestic program and ITER is discussed. The latter is seen from the perspective of industry support to the U.S. contribution to ITER, through the recently established ITER support office, but also from the more complex role that industry can play as the integrator and project manager of ITER, regardless of where it is constructed.

To support the U.S contribution, U.S. industry must be prepared to provide the services and materials/equipment/components that the U.S will commit to provide as either in-kind contributions or pay for as part of their share of the ITER final design and construction. The presently contemplated contribution of the U.S, and the ability of industry to support those contributions are summarized.

ITER has a number of choices regarding how the entire project will be managed during the final design and construction phase. A number of activities that industrial experience has demonstrated as crucial for the success of major projects are rendered complicated by the international nature of the program. Detailed planning of the procurement, fabrication, inspection, delivery, assembly of the various components into systems, integration of the systems with the construction of the facilities, start-up and testing has only been done at a superficial level. Such project execution plan, which is an absolute necessity for ITER success would rely on established procurement systems, project planning and control systems. During the procurement cycle, expediting and vendor inspection, logistics, receipt inspection and warehousing play a vital role in assuring that the items are available when needed. Nevertheless the main leverage that the program manager normally has on the various contractors that supply the items and services will often be absent from ITER, since the majority of funding is likely to be controlled by the various national parties, and not by the central ITER organization. During the actual assembly and construction, changes will occur that will require decisions, and interaction with the various contractors affected by the changes, and these changes will have funding implications, with funding not in control of the overall program management.

How ITER may choose to use industry in solving these problems is discussed. Alternatives that range from ITER building the capability to do all of the management and integration in its central organization to delegating this responsibility to an industrial company or groups and the advantages/disadvantages of the multiple options are discussed. Finally a recommendation is made for what the author believes to be the most readily doable option.

The National Ignition Facility: Laser Performance and First Experiments

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Abstract

The National Ignition Facility (NIF) at the Lawrence Livermore National Laboratory is a stadium-sized facility containing a 192-beam, 1.8-Megajoule, 500-Terawatt, ultraviolet laser system together with a 10-meter diameter target chamber with room for nearly 100 experimental diagnostics. NIF will be the world's largest and most energetic laser experimental system, providing a scientific center to study inertial confinement fusion (ICF) and matter at extreme energy densities and pressures. NIF's energetic laser beams will compress fusion targets to conditions required for thermonuclear burn, liberating more energy than required to initiate the fusion reactions. Other NIF experiments will study physical processes at temperatures approaching 10^8 K and 10^{11} bar, conditions that exist naturally only in the interior of stars, planets and in nuclear weapons. NIF has successfully activated, commissioned and utilized the first four beams of the laser system to conduct over 300 shots between November 2002 and August 2004. In doing so, NIF laser scientists have established that the laser meets all performance requirements on a per beam basis for energy, uniformity, timing, and pulse shape. Using these four beams ICF and high-energy-density physics researchers have conducted a number of experimental campaigns, resulting in high quality data that could not be reached on any other laser system. This presentation discusses NIF's early light commissioning and performance program, NIF's current and future experimental capability, and potential enhancements to NIF such as green laser operation and high-energy short pulse operation.

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European Technological Effort in Preparation of ITER Construction

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Europe has started, since the years '80s with the preparatory work done on NET, the Next European Tokamak, the successor of JET, to prepare for the construction of the next generation experiment on the road to the fusion reactor. In 2000, the European Fusion Development Agreement (EFDA) has been signed by sixteen countries, including Switzerland, not a member of the Union. Now the signatory countries have increased to twenty-five. A vigorous programme of design and R&D in support of ITER construction has been conducted by EFDA through the coordinated effort of the national institutes and laboratories supported financially, in the framework of the VIth European Framework Research Programme (2002-2006), by contracts of association with EURATOM. In the last three years, with an expenditure of 160 M€, the accent has been particularly put on the preparation of the industrial manufacturing activities of components and systems for ITER. Prototypes and manufacturing methods have been developed in all the main critical areas of machine construction, with the objective of providing sound and effective solutions: vacuum vessel, toroidal field coils, poloidal field coils, remote handling equipment, plasma facing components and divertor components, electrical power supplies, generators and power supplies for the Heating and Current Drive Systems and other minor subsystems.

Europe feels to be ready to host ITER site and to provide adequate support and guidance for the success of construction, wherever needed, to our partners in the ITER collaboration.

Oral Session II-1

ARIES Compact Stellarator - Special Session

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Exploration of Compact Stellarators as Power Plants: Initial Results from ARIES-CS Study

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The optimum stellarator configuration is quite dependent on the engineering/technological constraints. In a stellarator, the majority of the confining field is provided by the external coils (poloidal field is generated by external coil as well as the bootstrap current). Because the external coils produce a multipolar field, the magnetic field intensity drops rapidly away from the coil. As such, the space used by the first wall, blanket, shield, *etc*. in a power plant plays a crucial role in determining the external coil design and physics configuration optimization. Fixed-boundary analysis of stellarator configuration may lead to high-performance plasma configuration which cannot be produced with any practical coils and/or cannot accommodate a power-producing blanket. Constraint imposed by magnet technology such as maximum bend radius, support structure, and inter-coil spacing needed for assembly as well as maintenance of in-vessel components play a critical role in configuration optimization. As a whole, there are a large number of tradeoffs among physics parameters and engineering constraints. These trade-offs play a crucial role in optimizing the compact stellarator configuration.

A detailed and integrated study of compact stellarator configurations, ARIES-CS, was initiated recently to advance our understanding of attractive compact stellarator power plants and to define key R&D areas. The stellarator configuration space is quite complex because of the large number of independent parameters (*e.g.,* β, α-particle loss, aspect ratio, number of periods, rotational transform, sheer, *etc.*). Furthermore, engineering requirements and constraints such as coil topologies and maintenance approaches (which will have a major impact on in-vessel components, blanket, and power systems) may depend on details of a specific configuration. As such, the study ARIES-CS is divided into three phases. The first phase of the study was devoted to initial exploration of physics and engineering options, requirements, and constraints. Several compact stellarator configurations such as quasi-axisymmetric and quasi-helical were considered. In each case, trade-offs among plasma parameters (*e.g.*, α-particle loss versus β) was explored and possible coil topologies (modular/TF/PF trade-offs) was studied. Initial estimates of device size, first-wall and blanket power loadings, divertor heat loads, *etc.* were made with a systems model. Promising configurations identified in phase 1 will be subjected to detailed self-consistent analysis and optimization. Detailed self-consistent analysis of this phase will allow us to identify critical high-leverage areas for compact stellarator research. One of the promising configurations chosen in this phase would be used for a point design study in phase 3.

We have completed phase of 1 of ARIES-CS study—our results are described in this paper. We have identified several promising stellarator configurations. The trade-offs among physics parameters have been explored by parametric system analysis. It appears that devices with an overall size similar to those envisioned for tokamak power plants are possible. Our examination of engineering options indicates that overall maintenance approach plays a critical role in identifying acceptable engineering paths and has a major impact on plasma dimensions and performance.

Reactors with Stellarator Stability and Tokamak Transport

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The discovery of quasiaxially symmetric stellarators (QAS), for which the spectrum of the magnetic field has approximate two-dimensional symmetry, opens up the possibility of designing fusion reactors that have tokamak confinement and stellarator stability. This may enable power plants to operate in a steady state at high β without disruptions. We have developed stellarator reactors that possess these characteristics and have two or three field periods. The configurations are compact, with aspect ratios ranging from as low as 3 up to as high as 6. The rotational transforms ι are in the ranges $0.55>\iota>0.4$ and $0.7>\iota>0.4$ for the two and three field period configurations, respectively. The asymmetric terms in the magnetic spectrum, which can be less than one percent, are almost as small as the coefficients for a typical tokamak that are associated with ripple from the toroidal coils or helical excursion of the magnetic axis resulting from MHD instability. Reactors yielding 1 GW of electric power appear to be possible with major radius ~7 m.

We have also developed preliminary designs of the coils that are required to shape the plasma for overall systems analysis and engineering studies. Solutions with only twelve coils have been found for the two field period configuration. These modular coils are only moderately twisted, producing robust flux surfaces that do not deteriorate when changes are made in the vertical and toroidal fields. Filaments specifying the coils have a distance from the separatrix exceeding 1.4 m for a reactor with R~7 m. There is adequate room for radiation shielding to protect the coils and for the breeding blanket to produce tritium.

Recent LHD and W7-AS experiments have observed high β values which exceeded the limit predicted by numerical solutions based on linear, ideal MHD theory. The problem may be that force balance and stability are lost across islands if the equilibrium equations are not in conservation form. In light of the new experimental results, we shall discuss nonlinear MHD stability analysis with the NSTAB code, which employs a conservation form of the magnetostatic equations to calculate weak solutions that capture discontinuities modeling effectively both current sheets and small chains of magnetic islands. The application of NSTAB computations to the design of stellarator reactors will be illustrated.

Optimization of Stellarator Reactor Parameters*

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Stellarators have the potential for an attractive, fully ignited reactor. They are inherently steady-state devices without the large driven plasma current of the tokamak and spherical torus approaches. This characteristic reduces both the power needed to sustain the plasma and the risk of damaging disruptions. However, earlier stellarator power plant studies led to large reactor sizes. The German HSR reactor study had an average major radius R = 22 m in the five-field-period (M = 5) embodiment and R = 18 m in the newer M = 4 version. The M = 4 ARIES Stellarator Power Plant Study reactor with R = 14 m was a first step toward a smaller size reactor. The recent development of the compact stellarator concept allows reactors with major radius closer to that of tokamak reactors.

There are a number of factors that determine stellarator reactor size, primary of which is the distance needed between the edge of the plasma and the nonplanar magnetic field coils for the plasma scrapeoff region, the first wall, the blanket and shield, the coil case, and assembly gaps. Stellarator coil configurations with a smaller plasma-coil distance lead to larger reactors and coil configurations with larger plasma-coil distance have more convoluted coils and higher maximum magnetic field Bmax on the coils, which reduces the maximum allowable field on axis, B0. Other considerations in determining the optimum reactor size are the minimum distance between coils, neutron and radiative power flux to the wall, and the beta limit.

Three tools have been developed to optimize the main reactor parameters (R, B0, cost, etc.). A 0-D code allows assessing the compatibility of different constraints for a given power output: plasma-coil spacing, coil-coil spacing, Bmax and coil current density, neutron wall loading, plasma beta value, etc. A 1-D power balance code is used to study the path to ignition and the effect of different plasma and confinement assumptions including density and temperature profiles, impurity density levels and peaking near the outside, confinement scaling, beta limits, alpha particle losses, etc. for a given plasma and coil configuration. A reactor systems/optimization code is used to optimize the reactor parameters for minimum cost of electricity subject to a large number of physics, engineering, materials, and reactor component constraints. Different 1-D transport models including self-consistent electric fields, reactor component models, and costing algorithms are used to test sensitivities to different models and assumptions.

Three different magnetic configurations were analyzed: an M = 3 NCSX-based plasma configuration (NCSX-R) with coils modified to allow a larger plasma-coil spacing, an M = 2 plasma configuration (MHH2) with coils that are closer to the plasma on the outboard side with less toroidal excursion, and a scaled version of the HSR plasma and coil configuration (HSR*). The NCSX-R configuration is compatible with sector maintenance while the MHH2 and HSR* configurations are more suited for maintenance through ports. The reactors have major radii R in the 6-8 m range with an improved blanket and shield concept and an advanced superconducting coil approach. The low recirculating power should make compact stellarator reactors cost competitive with tokamak reactors.

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Attractive Design Approaches for a Compact Stellarator Power Plant

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The ARIES-CS study has been launched with the goal of developing through physics and engineering optimization a more attractive power plant concept based on a compact stellarator (CS) configuration. On the physics side, the first phase of the study involves scoping out different physics configurations including two and three field period options. Key considerations impacting the design of the CS include the size of the reactor, access for maintenance and the minimum distance between plasma and coil that affects shielding and also breeding if sufficient blanket coverage is not provided.

The first-phase engineering effort, carried out in close interaction with the physics effort, has focused on scoping out maintenance schemes and blanket designs best suited to a CS configuration. This has been done by building on information and results from past studies. The results from this effort will enable a down-selection to a couple of most attractive combinations of blanket configuration and maintenance scheme for more detailed studies which will then culminate in the choice of a point design for a full system design study during the final phase of the study.

To provide a broad range of possibilities to accommodate the physics optimization of the number of coils and the machine size, three possible maintenance schemes were considered: (1) replacement of an integral unit based on a field-period including disassembly of the modular coil system; (2) replacement of blanket modules through maintenance ports arranged between each pair of adjacent modular coils; and (3) replacement of blanket modules through a limited number of designated maintenance ports. Several possible blanket/shield configurations compatible with the maintenance schemes and the CS geometry were considered, covering the following three classes: (1) self-cooled liquid metal blanket with SiC_f/SiC composite as structural material or with He-cooled ferritic steel (with or without thermal/electrical insulation); (2) He-cooled solid or liquid breeder blanket with ferritic steel; and (3) self-cooled flibe blanket with ferritic steel. The divertor heat load for a CS is still uncertain but it is likely that, for a conventional divertor, He-cooling will be needed. As guidelines for the first phase of the study, it was decided to develop each concept to an extent sufficient for a credible case to be made regarding performance, fabrication and maintenance.

This paper summarizes the results from this first-phase engineering effort, covering the different blanket configurations and maintenance schemes. The main design parameters are summarized and key issues are discussed including the impact of different physics configurations on the engineering choices. These results will be used as the basis to down-select to a couple of combinations of blanket configuration and maintenance scheme for more detailed studies.

Benefits of Radial Build Minimization and Requirements Imposed on ARIES Compact Stellarator Design

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After two decades of stellarator power plant studies, it was evident that a new design that reflects the advancements in physics and improvements in technology was needed. To realize this vision, the multi-institutional ARIES team has recently launched a study to provide perspective on the benefits of optimizing the physics and engineering characteristics of the so-called compact stellarator (CS) power plants. The primary goal of the study is to develop a more compact machine that retains the cost savings associated with the low recirculating power of stellarators, and benefits from the higher beta, smaller size, and higher power density, and hence lower cost of electricity, than was possible in earlier studies.

It is widely recognized among stellarator researchers that the minimum distance between the plasma boundary and the middle of the coil (Δ_{min}) is of great importance for stellarators as it impacts the machine parameters. Specifically, it controls the minimum major radius and the maximum field at the coil. Techniques for minimizing the radial build have made impressive progress during the first year of the ARIES-CS study. During this period, several blanket/shield systems have been examined: one solid breeder-based system (Li_4SiO_4/FS/Be/He) and four liquid breeder-based systems (Flibe/FS/Be, LiPb/SiC, LiPb/FS/He, and Li/FS/He). As predicted, each concept offers advantages and drawbacks and an integrated study with guidance from the economic analysis and maintenance scheme will later identify the preferred blanket/shield concept for ARIES-CS.

The limited space assigned for the internals (blanket, shield, and vacuum vessel) calls for a well optimized, highly compact radial build at Δ_{min} in particular. Our analysis indicates that the radial distance between the plasma and coil center varies widely with the proposed blanket concepts, ranging from 1.3 to 2 m. A novel approach has been developed for ARIES-CS where the blanket at the critical area surrounding Δ_{min} has been replaced by a highly efficient WC-based shield. As a result, an appreciable 20-30 cm savings in the radial build has been achieved, reducing the major radius by 15-20%, which is significant. This approach places a premium on the blanket that covers ~90% of the first wall area to supply all the tritium needed for plasma operation. The economic benefit of this approach is yet to be determined and the added engineering problems and complexity will be addressed during the remaining period of the study. Future work present some challenges: how to integrate the thinner WC-shield with the blanket system, determining if there is a need for a separate decay heat removal loop for the WC-shield, and how to handle the relatively massive WC modules. This paper covers the details of the radial build optimization process that contributed to the compactness of ARIES-CS and describes how the ARIES team has tackled the emerging design problems. Compared with previous designs, the major radius of ARIES-CS has more than halved, dropping from 24 m to less than 10 m, making a step forward toward the feasibility of a compact stellarator.

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Oral Session II-2

Target Development and IFE Technology

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Demonstrating a Target Supply for Inertial Fusion Energy*

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A central feature of an Inertial Fusion Energy (IFE) power plant is a target that has been compressed and heated to fusion conditions by the energy input of the driver. The technology to economically manufacture and then position cryogenic targets at chamber center is at the heart of future inertial fusion energy power plants and significant development programs are underway. For direct drive IFE (laser fusion), energy is applied directly to the surface of a spherical CH polymer capsule containing deuterium-tritium (DT) fusion fuel at approximately 18 K. For indirect drive (heavy ion fusion, HIF), the target consists of a similar fuel capsule within a cylindrical metal container or “hohlraum” which converts the incident driver energy into x-rays to implode the capsule. Either target must be accurately delivered to the target chamber center at 5–10 Hz with a precisely predicted target location. The relatively fragile cryogenic targets must survive injection into the target chamber without thermal damage. Future successful fabrication and injection systems must operate at the low cost required for energy production (about \$0.25/target, about 10^4 less than current costs).

Z-pinch driven IFE (ZFE) utilizes high current pulses to compress plasma to produce x-rays that indirectly heat a fusion capsule. ZFE target technologies utilize a repetition rate of approximately 0.1 Hz with a higher yield. The cryogenic targets for ZFE must survive the longer time frame required for placement at target chamber center. Despite the differences, there are important synergisms in target technology between these three proposed approaches to IFE.

This paper provides an overview of the proposed target methodologies for laser fusion, HIF, and ZFE, and summarizes advances in the unique materials science and technology development programs. Demonstrating a credible pathway to an economical target supply is a major part of establishing IFE as a viable energy source. Although much work remains to be done, initial results are promising and suggest that a reliable, consistent and economical target supply can be developed.

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Development of target fabrication and injection for Laser Fusion in Japan

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At ILE Osaka University, the FIREX-1 project (Fast Ignition Realization Experiment) is in progress aiming heating of compressed plasma to the ignition temperature 10 keV. Targets used in this experiment have been developed with collaboration of National Institute for Fusion Science (NIFS), Fukuoka University, Levedev Physical Institute through ISTC project and General Atomics through Japan-US collaboration. Injection and tracking of fast ignition (FI) target are studied at the Hiroshima University and the Gifu University.

Up to date, accuracy requirements for the sphericity, thickness uniformity of the layers of the FI target are not clear. They are, however, considered to be much relaxed than those for a central ignition target. In previous experiment at ILE, 600 times solid density was achieved using a deuterated, tritiated polystyrene shell while simulation on the hydrodynamic instability indicated that the imploding shell broke before it reached its final stagnation[1]. The FI target would accept certain low and middle mode non-sphericity. Figure 1(a) illustrates the fast ignition target for FIREX-1. The target consists of a low- density foam-layer saturated with solid deuterium and tritium (DT), a gas barrier, and a guide cone for additional heating laser. The fuel is fed through a capillary tube from a reservoir located outside the target chamber. This scheme reduces the burden related to use high pressure DT gas.

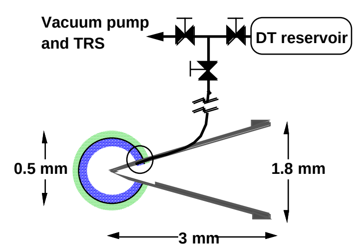
Figure 1(b) is designed for a reactor. This target has a thermal insulator outside and a heavy cone whose inner surface is paraboloid to help focusing of the additional heating laser to the compressed core.

Technical challenges to make these targets are; to make low density foam (10mg/cc), to make a hole for cone assembling on a fragile shell, to make a thin capillary, disassembling of parts due to different thermal expansion coefficient at low temperature and characterization of solid DT layer.

A cryostat to demonstrate proof of principle of foam method for the cone target is now being assembled at NIFS.

At the Hiroshima University, pneumatic injection with rifling is studied and the tracking of FI target with cone was developed at the Gifu University using matched filter method that can eliminate the image processing during detection.

Detail of these activities will be reported at the meeting.



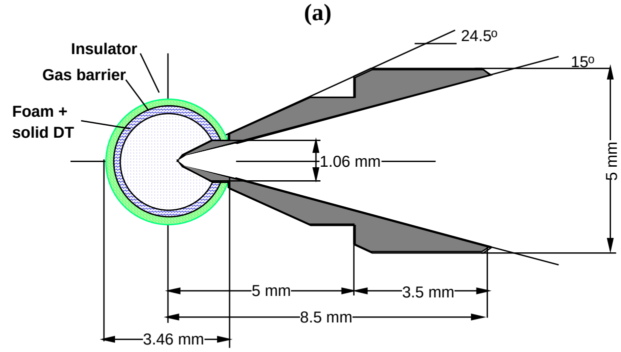


Fig. 1 Cone guided FI target for FIRX-1 (a) and a reactor (b)

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High Energy Density Simulations for Inertial Fusion Energy Reactor Design

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The promise of inertial fusion as an energy source for electric power generation is elucidated in conceptual reactor design studies. These studies use the best available experimental information to predict the fusion performance and performance of the other reactor components such as materials strength and radiation damage, tritium breeding heat transfer, etc. For those properties where experimental evidence is not available, computer simulations are often used. One aspect where computer simulations are predominantly relied upon are the high energy density radiation hydrodynamics simulations of target burn and expansion and the reactor cavity and first wall response to the energetic target constituents (x-rays, ions, and neutrons).

Simulations of target implosion and ignition are of course a well developed discipline and are a cornerstone of the U.S.D.O.E. inertial confinement fusion program and other programs worldwide. The post-ignition target performance and subsequent target expansion are less well studied and are the subject of this presentation. The starting point is the same radiation hydrodynamics model used to simulate target implosion and burn. We will explore the possibility that this fluid-based model is inadequate to simulate target expansion in a vacuum. We will report on the implications for reactor design and performance of an imprecise estimate of the nature of the energetic target constituents.

Targets for Heavy Ion Inertial Fusion Energy*

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In the past few years, the emphasis for heavy ion targets has moved from the "distributed radiator" target [1,2] to the "hybrid" target [3] because the hybrid target allows a large beam spot. Since focusing the ion beam to a small spot is difficult, the hybrid target should ease the constraints on the accelerator and make the accelerator less expensive. Since the cost of the accelerator is the largest item in a heavy ion inertial fusion power plant, reducing the cost of the accelerator should reduce the cost of the plant and the ultimate cost of electricity.

The hybrid target introduces some new target physics issues, however. In particular, the symmetry required by the capsule is controlled using internal shields inside the hohlraum, as opposed to using proper placement of the beams. The latter technique is used in the heavy ion distributed radiator and the National Ignition Facility point design. In the hybrid target, we use a shine shield to prevent the capsule from "seeing" the spot where the ions beams deposit their energy. Without the shine shield, the capsule would see a large P_2 (second Legendre polynomial) asymmetry. Radiation flows around this shine shield onto the capsule. This results in a significant P_4 asymmetry. Our solution has been to correct this P_4 by using a shim—a thin layer of material placed on or near the capsule surface to remove the small amount of excess radiation.

The double-ended Z-pinch target [4,5] has many similarities to the heavy ion hybrid target. It also uses a shine shield to block radiation from the radiating z-pinch. Again, radiation flows around this shine shield and can result in a large P_4 asymmetry. The similarities between these targets has led to a collaboration on exploring shims to control symmetry.

Our first set of experiments used a shimmed capsule to remove a P_2 asymmetry in a double-ended Z-pinch. A layer of gold, with thickness that varied as a function of polar angle, was deposited on the capsules by GA. The first set of experiments were successful in reversing the sign of the P_2 asymmetry and shows that we can control the asymmetry by using this technique.

In this talk, we will discuss the hybrid target, its impact on the heavy ion fusion power plant, as well as the design, target fabrication, and experiments to test shims for symmetry control.

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Fabrication of Overcoated Divinylbenzene (DVB) Shells

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The High Average Power Laser Program is a multi-lab effort to develop an Inertial Fusion Energy (IFE) reactor for electrical production. The driver would consist of modular Krypton Fluoride or Diode Pumped Solid State Lasers, direct drive targets, and dry wall chambers. Target models for this driver require a 4 mm diameter capsule with a core of deuterium tritium (DT) vapor and layer of DT ice, a layer of low density foam, filled with DT ice, a full density polymer overcoat, and a high-Z metallic overcoat. Requirements of the foam are a density from 30–140 mg/cm^3, cell size less than 5 microns, and possibly only containing carbon and hydrogen. Divinylbenzene was developed as a foam system to meet these requirements as the system can be produced within the density range of interest, has a cell size of 1-4 μm, and contains no oxygen. The ongoing development of DVB shells to meet the specifications of an IFE reactor is presented. The shells are formed using a water-oil-water emulsion microencapsulation technique and are agitated during thermal gelation to increase shell concentricity. An interfacial polycondensation technique using poly(vinyl phenol) reacted with an acid chloride was selected as a straightforward method to apply the full density overcoat. Research specification goals are as follows: 4 mm diameter, 300 micron foam wall thickness, 100 mg/cm^3 foam wall density, <1% wall nonconcentricity, 1-5 micron overcoat thickness, and overcoat surface roughness <20 nm RMS. Diameter, wall thickness, density, and overcoat thickness requirements have been met. Two percent is the lowest nonconcentricity achieved for an individual shell; four percent is the lowest for a batch. 50 nm RMS surface roughness is the minimum that has been achieved. Formation, gelation, characterization, overcoating techniques, and methods to further reduce nonconcentricity and surface roughness are presented. Concepts to scale-up these processes are also presented as approximately 500,000 targets per day would be required by an operational IFE reactor.

Work performed under the auspices of the Naval Research Laboratory under contract **N00173-03-C-2023.**

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Oral Session II-3

Latest Technology and Tritium System

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Recent technological progress for Advanced Tokamak Research in JT-60U and JFT-2M

N. Hosogane, the JT-60 team and JFT-2M team

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Japan Atomic Energy Research Institute has been investigating an advanced tokamak concept for ITER, and will lead it to an economically and environmentally acceptable future fusion reactor. In JT-60U, to develop such a reactor with high output density, advanced tokamak operations based on high β plasmas with a high bootstrap current fraction have been studied. In JFT-2M, on the other hand, "advanced material test experiment"(AMTEX), which investigates the feasibility of ferromagnetic ferritic steel with low activation property to tokamaks as a future reactor material, has been conducted from a viewpoint of reducing radioactive wastes of fusion reactors.

From 2003 experimental campaign, JT-60U proceeded to a new stage of steady state study with long discharges exceeding time scales of current diffusion and wall saturation. For this purpose, the tokamak discharge duration was extended from 15 s to 65 s by modifying the control systems of power supplies. Also, to sustain high performance plasmas in the steady state with NBI and RF systems, intensive efforts have been made to extend their injection pulse lengths to 30 s or more.

(1) Positive ion based NBI system

This system was extended to 30 s at 14 MW only by modifying the control system. The pulse length is mainly limited by temperature rise at the injection port without water cooling.

(2) Negative ion based NBI system

The electric field at the beam extractor was corrected so as to compensate the deflection of the beamlets caused by repulsion between beamlets. In addition, pumping outlets were added at either side of the accelerating grids to reduce bombardment of electrons stripped from negative ions due to collisions with neutrals. These reduced the temperature rise at the injection port by 70% and the heat load to the grounded grid by 25%, showing a good prospect to the achievement of 30 s at 2 MW. The pulse length has reached 17 s at 1.6 MW and 366 kV to date.

(3) Electron cyclotron heating system

The heater current and anode voltage of a gyrotron were actively controlled during or just before the operation to keep the resonance condition, and cooling of the transmission components were improved. Using a waveguide-type dummy load with an absorption capacity of 1 MW, continuous operation for 16 s at 0.4 MW has been achieved for one unit.

(4) Lower hybrid heating system

To prevent the stainless steel antenna mouth from severe damage due to RF breakdown and heat load from plasma, the heat resistive carbon-grill-antenna was attached to the existing antenna base. In the aging process, RF breakdown is still obstacle to increasing the LHRF power, but no abnormal temperature increase has been observed.

AMTEX in JFT-2M started in 1996, and finished in March, 2004. The feasibility of ferritic steel to tokamaks has already been shown with respect to compatibility with plasma equilibrium control, confinement characteristics such as H-mode, etc. In the last campaign, the influence to high β plasmas such as β-limit etc., which are crucial to advanced tokamaks, was investigated. The result is under analysis, and will be presented together with the previous ones.

An Integrated Approach To Fusion Material Research At SCK•CEN

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For the last 20 years, fusion material programs in Europe, Japan and US have been focused on developing Reduced Activation Ferritic/Martensitic (RAFM) steels as prominent structural materials, which would reduce the environmental impact of the irradiated steel after the service lifetime of a fusion reactor.

In the European Union, within the Long Term Programme of EFDA (European Fusion Development Agreement), remarkable efforts are spent by several scientific institutions for the characterization and optimization of the European reference RAFM steel, EUROFER97. This was modelled after the conventional T91 alloy and exhibits a tempered martensitic microstructure which allows operation at relatively high temperatures (up to 500 – 550 °C).

Within the Belgian Nuclear Centre (SCK•CEN), an integrated approach to the characterization of EUROFER97 is being consistently applied; this includes the following investigations, which are conducted in parallel and whenever possible in synergy.

- *Neutron irradiations in the BR2 reactor*. Several irradiation experiments on EUROFER97 have been performed in the period 2000-2003, all at 300 °C and corresponding to accumulated doses in the range 0.3 to 2 dpa. Further irradiations are foreseen in 2004-2005.
- *Characterization of the unirradiated and irradiated mechanical properties.* Tensile, instrumented Charpy and static fracture toughness tests have been performed in the baseline condition and for different dose levels, allowing to assess the evolution of the mechanical properties of the material as a function of accumulated fluence (specifically, hardening and embrittlement). Comparisons with similar data for other RAFM steels are being made, and extensions to EUROFER97 welded joints and ODS materials are in preparation.
- *Investigation of environmentally assisted cracking (EAC).* This is a critical concern for the design of nuclear systems; for fusion reactor applications, there are two potentially corrosive environments: water at high temperature and liquid lead-lithium eutectic alloys, depending on the cooling option for the blanket. Furthermore, an interaction between radiation damage and EAC is likely. For both EAC and liquid metal embrittlement, the yield stress of the material is a known key factor. At SCK•CEN, a research programme is presently being carried out to investigate the influence of irradiation damage on both EAC and embrittlement in Pb-Li alloys; the first results of EAC tests on irradiated EUROFER97 are now available.
- *Multiscale modelling of radiation effects and specific effects on Fe-Cr systems.* The objective here is to model radiation effects in RAFM steels under fusion relevant conditions, in the range T = RT – 550 °C and in the presence of high concentration of irradiation induced impurities (H, He, etc). The applied methods range from atomic level ab initio calculations (MD – Molecular Dynamics and KMC – Kinetic Monte Carlo) up to mesoscopic dislocation dynamics and finite element at the macroscopic level. More specifically, work is in progress for the setting up of libraries of displacement cascades in Fe and Fe-Cr alloys using MD with an adequate interatomic potential. In parallel, multiscale models are developed capable of simulating, by means of suitable computational models, the evolution of radiation-induced damage in fusion-relevant model materials (e.g. Fe-Cr as model alloy for RAFM steel).

This paper will provide a general overview of the above mentioned investigations, as well as highlights of the most significant results obtained in the different fields of activity.

Comparison of Tritium Component Failure Rate Data

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Experimental fusion facilities have used, and others will use, tritium fuel. Because tritium is a radioactive fuel with unusual characteristics, such as its ability to become water, combustibility, and its ability to permeate many materials, there are safety issues with its storage and use. A number of experimental tritium laboratories have been built and operated to answer fusion-specific questions about tritium, including storage, isotope separation, and handling. Several of these tritium laboratories, including the Tritium Systems Test Assembly (TSTA) in the US, the Tritium Process Laboratory (TPL) in Japan, and the Tritium Laboratory Karlsruhe (TLK) in Germany, have recorded the component failures that have occurred. With these data sets and the operations information from each facility, component failure rates have been calculated and published. These failure rates are used in fusion safety assessment to model similar tritium systems and estimate the rate of adverse events, such as leaks. These data can also be used for estimating tritium system availability; that is, to model detrimental system events such as failure to provide tritium when required.

TSTA failure rates were first published in the late 1980's and early 1990's. The TSTA failure event data reports through the mid-1990's have also been collected and used to update the initial component failure rates. The failure event reporting was discontinued in the mid 1990's as the TSTA emphasis shifted after exhibiting over a decade of safe operations and the facility experienced reductions of funding. The TSTA was closed and decommissioned in the early 2000's, so this is the final update of the TSTA component failure rates. The other tritium facilities are continuing operation and will be able to periodically update their failure rate data values.

For this paper, the updated TSTA data are compared to the published failure rate data from the TPL, the TLK, and the Joint European Torus Active Gas Handling System. This comparison is on a limited set of components, but there are a variety of data sets to compare. The most reasonable failure rate values are recommended for use on next generation tritium handling system components, such as those in the tritium plant systems for the International Thermonuclear Experimental Reactor (ITER) and the tritium fuel systems of inertial fusion facilities. These data and the comparison results are also shared with the International Energy Agency cooperative task on fusion component failure rate data.

The data sets compared here generally show favorable agreement between values. Consensus failure rate values are recommended in this paper for some types of components. The variety of data available for comparison shows that these data are important for safety since several countries have devoted ongoing effort to collect the data and analyze it, and the variety of data also allows consensus, or average, values to be reached for some components. These consensus values can be used for safety assessment of next-generation fusion experiments.

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Developments in Remote Collaboration and Distributed Computing

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The National Fusion Collaboratory Project is creating and deploying collaborative software tools to unite fusion research in the U.S. In particular, the NFC is developing and deploying a national FES "Grid" (FusionGrid) that is a system for secure sharing of computation, visualization, and data resources over the Internet. The goal of FusionGrid is to allow scientists at remote sites to participate as fully in experiments, machine design, and computational activities as if they were working onsite thereby creating a unified virtual organization of the geographically dispersed US fusion community. To ameliorate the problem of divergent security systems, a unified secure remote computing middleware package, the Globus Toolkit, is used for secure communication, authentication, and remote resource usage.

The open-source Access Grid (AG) software is used by FusionGrid to create a service that enables secure group-to-group interaction and collaboration that improves the user experience beyond teleconferencing. It also allows for application sharing, so researchers at remote locations can simultaneously see the same graphical visualization. The AG system is being used for seminars, working meetings, and experimental operations. Most recently the AG service was used by a San Diego-based scientist to lead an experiment on JET.

In addition to remote collaboration, FusionGrid provides fusion researchers with computational services. The first computational service on FusionGrid is the TRANSP transport analysis code. Instead of using locally-installed software, researchers dispatch the TRANSP code for execution on FusionGrid. The security features of Globus make it possible for computing resources to be shared without fear of unauthorized use. A secure version of the MDSplus data storage system was developed to enable data sharing in a standardized format.

The SCIRun visualization software is used to visualize in 3D large datasets stored in MDSplus and HDF5. SCIRun provides a visual programming interface, which allows for rapid prototyping of new data models.

KEYWORDS: National Fusion Collaboratory, remote collaboration, grid computing

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Information Technology Systems for Fusion Industry and ITER project

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The industrial developments in fusion industry will have to overcome numerous technical challenges and will have strong needs for modern Information Technology (IT) systems. High-quality IT systems are an essential component to assure rapid scientific and technological progress as they can substantially accelerate the learning process and shorten the time needed for a development.

Fusion is based on data-intensive scientific applications and technologies, and will benefit significantly from the use of the modern Information Technologies and Techniques such as:

- Advanced data acquisition, data and signal processing analysis;
- Grid enabled information and computing infrastructure with database and data warehouses systems for storing large amounts of data;
- Automated data management with data fusion as a component;
- Problem solving environments that support data exploration, data mining, simulation modeling, and advanced query processing;
- High speed networking;
- Visualization means to intensify visual analysis of large datasets, support operations control systems, and decision support data systems;
- Near-line archiving and others.

A fusion industry manifested itself with an unprecedented international collaboration – International Thermonuclear Experimental Reactor (ITER). Data accumulated in ITER will be the major output of the project and will create the knowledge base for a future Fusion Power Plant. A modern and effective information infrastructure will be critical to the success of the ITER project.

To accumulate and maintain the knowledge base at all stages of the project, we propose to build an integrated Information System for ITER. We refer to this system as ITER Information Plant (IIP). IIP will be used as a high-quality tool to minimize lost experimental time and accelerate the understanding, interpretation, and planning of ongoing fusion experiments. ITER IP will allow reaping maximum benefits from the scientific and technological achievements of the project as it makes the ITER results accessible to hundreds of researchers worldwide. This is expected to dramatically increase the pace of scientific and technological discovery and the rate at which practical use is made of these discoveries.

The development of ITER Information Plant will require join expertise in fusion and information technology. We believe that it will benefit from international collaboration, as it allows bringing diverse technical expertise and creates a balanced and creative international environment.

Being a first of his kind, the ITER Information Plant could be used in the future as a prototype of the IT system for national and international fusion projects.

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Poster Session II

Plasma Control and Diagnostics
Non-Electric Applications and IEC Research
Safety and Environment
High Flux Components and Chamber Clearing
Thermal and MHD Analyses

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Magnetic method To characterize the Current Densities in a Breaker Arc

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The purpose of this research was to use magnetic induction measurements from a low voltage electric arc, to reconstitute the arc's current density. The measurements were made using Hall effect sensors, which were placed close to, but outside the breaking device. The arc was modelled as a rectangular current sheet, composed of a mix of threadlike current segments and with a current density varying across the propagation direction. We found the magnetic induction of the arc is a convolution product of the current density, and a function depending on the breaker geometry and arc model. Using deconvolution methods, the current density in the electric arc was determined.

The first experimental verification of the model was obtained using reconstruction of the current density in a conductor bridging the arc gap. Next, the method was used to study the arc behavior into the breaker device. Notably, position, arc size, and electric conductivity could all be determined, and then used to estimate an arc temperature. From the calculated current density and arc size, we could also characterize the arc mode, diffuse or concentrated, and study the condition of its mode changing.

Automatic Fault-Checking System on the DIII–D Tokamak*

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Modern tokamaks are highly sophisticated devices consisting of a large number of state-of-the-art systems that must function in unison to obtain a successful plasma discharge. An unsuccessful discharge can result if one or more systems fail, and may be very difficult to diagnose in an efficient and timely manner. The resulting reduction in tokamak availability and productivity can be expensive, therefore justifying a significant effort in the automation of fault diagnosis.

For the DIII–D tokamak, a software system has been developed to automatically monitor and test the performance of hundreds of tokamak systems and components. The Fault Identification and Communication System (FICS) is automatically triggered to run immediately after each tokamak discharge and report its results via a simple color-coded graphical user interface.

FICS is based on the C Language Integrated Production System (CLIPS), inference engine software in the public domain that was developed by the Software Technology Branch at NASA. The execution of FICS is driven by the availability of data following a tokamak discharge, therefore resulting in minimal delay before the start of processing. The data-driven feature also makes it relatively straightforward to add new rules, with no need to modify the logic structure, keeping FICS current with changes in device operation.

The FICS automatic fault-checking system has been in routine use on the DIII–D tokamak since 1999. The system has expanded considerably since its inception and now performs routine tests on a variety of systems including power systems, computer systems, magnetic field coils, vacuum systems, gas injectors, plasma diagnostics, plasma shape control, auxiliary heating systems and even other fault detection systems. The diagnosis of an obvious fault after a discharge is often performed more quickly by an experienced operator, but FICS detects secondary faults that the human operator misses. The large number of routine tests run by FICS far exceeds the capabilities of the operator, allowing them to concentrate on other tasks. The significant advantage of FICS, however, is in its detection of insipient faults, e.g. the slow degradation of performance of a tokamak system that would have caused future discharges to fail. It has been estimated that FICS has saved an average of one to two shots per day, which equates to approximately 5% of all tokamak pulses. Examples and details of the FICS fault-detection system will be presented.

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Four Barrel Pellet Injector Upgrade on the Madison Symmetric Torus (MST)

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A two-barrel cryogenic pellet injector, installed in 2002, has recently been upgraded to allow up to four pellets of hydrogenic ice to be injected into a single MST plasma discharge. The injector system, based on a pipe gun configuration, was designed and fabricated by Oak Ridge National Laboratory (ORNL) and installed on an existing MST support structure. The gun, which was previously outfitted with two barrels, has been refitted with its full design complement of four barrels, producing pellets of diameters ranging from 1 mm to 1.6 mm. Pellet speeds are selectable and range from ~200 m/s to over 1000 m/s by the choice of a mechanical punch, a high-pressure propellant gas, or a combination of the two. Additionally, a new short-barrel geometry has been developed to allow reliable pellet speeds in the 500 m/s range, which is critical for MST type plasma parameters. To diagnose the pellets' speed and size, all four barrels share a common microwave mass detector, and each barrel is equipped with a light gate and photography station. The mass detector cavity geometry has also been optimized for MST-sized pellets.

Pellets are radially injected at an angle of 30 degrees from the outboard midplane through a newly installed set of ports on the MST vacuum vessel. Aligned about the plasma center, these new ports allow all four barrels a line-of-sight trajectory to the discharge core. The system uses a closed loop helium gas cryocooler for pellet formation, eliminating the inconvenience, expense and hazards of liquid cryogens. The entire system is self-contained and is controlled through a single, remotely sited instrumentation rack, which contains all of the necessary control and data acquisition electronics, including the control computer.

Initial results from the previous two-barrel gun show that during improved confinement plasmas, increases in line average density of nearly 50 percent persisting for several milliseconds have been achieved, consistent with improved particle confinement. With the full complement of four barrels, more comprehensive studies of edge and core fueling will be conducted, using fast and slow pellets of various sizes.

In addition to the standard complement of MST diagnostics used with pellet injection (fast CCD camera, single-chord CO_2 interferometer, multi-chord far-infrared interferometer and Thomson scattering,) a dedicated high-resolution spectrometer and a bichromator, fabricated by ORNL, will be installed for measurements of the deuterium pellet ablation cloud. High-resolution measurements of the Stark broadening of the D_α line will provide information about the pellet cloud density, while bichromator measurements of the D_α-to-D_β line ratio will yield information on the cloud temperature. These measurements, combined with pellet cloud images, will be crucial in determining if pellet ablation physics in the RFP is similar to that of other toroidal confinement devices.

Hybrid AL/SiC composite optics for IFE applications.

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The optical components for IFE optics places stringent requirement on the material selection. The surface reflectivity requirement limits the top surface selection to Aluminum. However, thermal stability of Aluminum optics greatly reduces its size. A novel approach to solve this problem is investigated by exploring the hybrid Aluminum / silicon carbine composite structures. The laser threshold damage under short-pulse laser condition of this new class of material will be discussed.

EBW coupling using a twin waveguide launcher on the MST reversed field pinch

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Electron Bernstein wave (EBW) injection is considered a promising method for localized heating and current drive overdense plasmas such as Reversed Field Pinches (RFPs). Theoretical work indicates that a finite perpendicular launch angle can improve coupling of externally launched electromagnetic waves, and that the coupling is predicted to be non-symmetric with respect to the launch angle. Experiments are reported here in which EBWs were launched by a twin waveguide antenna on MST. The reflection coefficients and phase shifts have been measured for each arm as the phase between waveguides is varied: for appropriate phasings, over 80% of the power can be coupled. In addition, density profiles at the edge are measured using an array of Langmuir probes, and this data is used to compute the reflection coefficients predicted by theory. Coupling measurements are consistent with theoretical models. Results are presented for a scan of the polarization (O and X modes) facilitated by a rotating antenna support.

Effect of Nuclear Elastic Scattering on Neutral Beam Injection Heating in Thermonuclear Plasmas

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An intense neutral beam injected into thermonuclear plasmas plays important role in various stages of fusion reactor operations. The beam particles slow down, deposit most of their energy via Coulombic collisions, and create a tail (non-Maxwellian component) in velocity distribution function of the same ion species as the one injected. It is well known that for suprathermal ions, the Nuclear Elastic Scattering (NES) by thermal ions contributes to the slowing-down process. According to the recent scaling up of the fusion experimental devices, the use of beam energy more than 1 MeV is considered. In this case, the NES effects on the slowing down process of injected beam particles may be important to understand device performance during plasma heating operations. The purpose of this paper is to quantitatively estimate the NES effect on neutral-beam-injection (NBI) plasma heating.

The NES is a non-Coulombic, large-energy-transfer (LET) scattering process. Devany & Stein[1] first pointed out the necessity of taking account the contribution of the nuclear-forces, including their interference with the Coulomb process, to ion-ion scattering, and presented an expression for the energy loss rate of fast ions due to the NES. In their expression, however, thermal motion of the background ions was neglected (background ions were assumed to be stationary), thus effect of the large-angle nuclear collisions which scatter the slowing-down ion itself up to the higher energy range, i.e. LET up-scattering effect, is not incorporated. Recently we have derived the energy loss rate of high-energy ions due to the NES, including the LET knocking-up of background ions from thermal to higher energy range[2]. On the basis of the derived expression, we have estimated the NES effect on the fraction of beam energy deposited to ions. In this paper we further estimate the NES effect on the penetration length of injected beam and the in-flight fusion reaction rate coefficient of suprathermal beam ions. It is shown that when the beam energy is higher than 1 MeV, the NES effects on NBI heating becomes appreciable.

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Integrated Plasma Control in Next-Generation Devices Using DIII–D Modeling and Simulation Approaches*

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An extensive set of software tools for integrated plasma control has been developed, validated, and applied to the DIII–D tokamak to design controllers for plasma shape and both axisymmetric and non-axisymmetric MHD instabilities [1]. The integrated plasma control approach uses validated physics models to design controllers, and confirms their performance by operating actual machine control hardware and software against detailed tokamak system simulations. These tools are being applied to several fusion device designs which require integrated plasma control for high reliability advanced tokamak or burning plasma operation, including KSTAR, EAST, and ITER. DIII–D physics-based models include conductors, diagnostics, power supplies, and both linear and nonlinear plasma models. Controller designs are based on multivariable linear methods and are robust to model uncertainty in dynamic shaping performance. Simulations incorporating detailed models are performed offline to confirm performance in the presence of such nonlinear effects as voltage saturation, power supply rate limits, and plasma configuration changes (e.g. from single to double-null). Plasma control systems (PCS) based on the DIII–D PCS [2] have been designed for each of these devices. These systems can be connected to the detailed control simulations to verify event handling and demonstrate functioning of control action under realistic hardware (cpu and network) conditions. Results of simulations are shown, illustrating limitations on performance imposed by each device design, engineering choices, and control system algorithms and hardware. Such simulations allow confirmation of performance prior to actual implementation on an operating device.

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Long pulse operation of NBI systems for JT-60U

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In JT-60U, the experiment for plasma discharge duration up to 65 sec has been conducted since last year so as to investigate the behavior of steady state plasma. To study high performance plasma for continuous discharge, it is necessary to extend pulse duration of neutral beam injector (NBI) from 10 sec (the current pulse duration limit); because the NBI system is main heating and current drive system. The four units, which tangentially inject neutral beam to plasma, were modified for long pulse operation. Modification on various electric power supplies, control system and beam limiter equipped with injection port was carried out.

In NBI system, one of the pulse duration limits is the temperature rise of the beam limiter. The limiter prevents damage of the injection port. The limiter is made of molybdenum and is not cooled actively. The temperature rise was observed to be 190 degrees for 7.6sec with 2 MW of NBI power per a unit. A three-dimensional solid model was constructed for a thermal Finite Element Model analysis to evaluate the temperature rise of the limiter. It was found that the maximum temperature of the limiter exceeded 750 degree for 30 sec beam injection. Therefore, a new limiter was designed to suppress the temperature rise. It has a shallow angle against beam and large volume.

On the other hand, another operation limit of NBI is re-ionization in the injection port. There is significant re-ionization of the neutral beam due to collisions with gas molecules, and then the re-ionized power may damage components in the injection port. Therefore the beam pulse duration was gradually expanded under checking the temperature of the limiter and the pressure in the port. Up to now, it was attained successfully that the injection pulse length was extended up to 30 sec with 2MW power at 80keV by one unit. 330MJ of the total integrated injection beam power into JT-60U plasma was achieved, including the negative-ion based NBI operation for 17 sec.

Nonlinear Methods for Current Limit Constraint Satisfaction in Tokamak Plasma Shape Control*

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The DIII–D tokamak is capable of supporting a wide variety of plasma equilibria because of its relatively large number of coils and their proximity to the plasma. To support its advanced tokamak mission, the DIII–D experimental program continues to push the envelope of this capability, frequently encountering limits imposed by allowable currents in poloidal shaping coils.

With standard DIII–D control algorithms based on approximately one coil controlling each boundary control point, violation of these current constraints is currently dealt with by operator adjustment of control targets and gains between plasma discharges. Accuracy in control is sometimes sacrificed for adherence to current limits in these highly tuned controllers, since violation of a current limit usually causes a premature end to the plasma discharge.

Demands for more precise and stable control have motivated efforts to develop and install advanced multivariable algorithms for control of plasma shape in DIII–D and other devices. Although various linear minimization schemes can be implemented to encourage currents to remain within limits, adherence to these limits cannot be guaranteed by linear methods alone. This limitation has been clearly observed in the process of implementing linear multivariable controllers on DIII–D. There is currently no way to ensure respect of nonlinear current constraints in a multivariable linear controller design and no practical way to manually tune these fully coupled controllers between discharges after installation.

In this paper, we discuss ongoing efforts to provide methods which guarantee currents will not exceed preset limits, and to simultaneously achieve the best obtainable quality of control subject to the current limit constraints. These methods are necessarily nonlinear and include both constrained and unconstrained minimization algorithms, calculated in real time. The methods developed are generalized to the multiple control circuit configurations and to the multiple plasma equilibria supported by DIII–D. Results of experimental implementations will be described.

These methods have important applications in future devices, since the ability to design and build devices with smaller control margins can mean a significant savings in cost of construction and operation.

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Nonrigid, Linear Plasma Response Model Based on Perturbed Equilibria for Axisymmetric Tokamak Control Design*

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Tokamak control design relies on an accurate linear model of the plasma response, which can often dominate the local field variations in regions which affect equilibrium parameters under active feedback control. For example, when fluxes at selected points on the plasma boundary are regulated in DIII-D, the plasma response to a change in a coil current gives rise to a flux change which can be larger than and different from the flux change caused by the coil alone.

In the past, rigid plasma models have been used for linear stability and shape control design. In a rigid model the plasma current profile is considered fixed and moves rigidly in response to control coils to maintain radial and vertical force balance. In a nonrigid model, however, changes in the plasma shape and current profile are taken into account. Such models are expected to be important for future advanced tokamak control design.

The plasma equilibrium is described by the Grad-Shafranov equation, $j_\phi = Rp' + FF'/\mu_0 R$, where j_ϕ is the toroidal plasma current and p', FF' are stream functions of the poloidal flux which depend on currents in coils surrounding the plasma and on kinetic and transport characteristics of the plasma. In the absence of kinetic and transport input, one must make a closure assumption to produce a consistent model. One approach to such closure is to make assumptions about the perturbational behavior of the two stream functions based on fundamental physics constraints and supported by empirical observations. We can for example apply the simple assumption that the stream functions are constants with respect to the flux on the magnetic axis so that $p'(\psi - \psi_{axis})$, $FF'(\psi - \psi_{axis})$ are unaffected by plasma perturbations. Other constraints can be applied, for example representing fluid element current conservation. Previous studies have shown that exact flux conservation does not accurately represent the plasma response on control-relevant timescales, and that including the resistive plasma response is important in order to reproduce experimental data (e.g. [1]).

Development of a nonrigid plasma response model for high-accuracy multivariable control design will be described, and comparisons with rigid model predictions will be shown along with validation of model responses against DIII-D experimental data. The linear perturbed plasma response model is calculated rapidly from an existing equilibrium solution without the need for explicit solution of elliptic partial differential equations such as a perturbed Grad-Shafranov equation.

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Numerical Study on behavior of hydrogen ice pellet in drift tube

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A fuelling method using a bent drift tube to guide hydrogen ice pellets into fusion plasma is proposed as a flexible one that can have a variety of injection locations and injection angles in plasma. During passing through inside the bent tube, ice pellet is considered to undergo not only friction (or contact) with tube wall but radiation from the wall. This leads to mass deficit of the ice pellet and consequently drastically affects plasma properties such as density and temperature profiles. In order to predict the effect of fuelling the plasma with pellets in a fusion reactor, it is of vital importance to have a pellet ablation model that adequately describes the experimental data.

In this study, we focused on behaviors of hydrogen ice pellet in the vicinity of tube wall, especially on thermal contact conductance and friction with the wall. We numerically estimated contributions of these phenomena to the mass deficit of the pellet. Material Point Method was adopted to simulate motions of the pellet and wall, and Smoothed Particle Hydrodynamics method was for thermal field.

We examined various cases with varying initial velocities of ice pellet, injection angles and coefficient of kinetic frictions as numerical parameters. It was clarified our simulations that when there was no friction between ice pellet and wall, mass deficit tended to decrease as injection angle became large. This is because large injection angle made a contact interval short and input heat from wall to pellet became small. In friction cases, mass deficit became obviously large compared with the no friction cases. In these cases, contribution of frictional heat to mass deficit was dominant compared with that of thermal contact conductance. Finally, we obtained the following correlation equation;

$$\Delta m = 7.65\mathrm{x}10^{-5}\, V^{2.1}\, (\sin\theta)\, \mu^{1.4} \tag{1}$$

where Δm and V correspond to the ratio of mass deficit and the initial velocity of ice pellet, respectively while θ and μ are injection angle and coefficient of kinetic friction, respectively.

We also examined the decrease rate of ice pellet velocity before and after impact. It was found that the decrease rate became large as large injection angle and large kinetic friction coefficient. It was also clarified from our calculations that the decrease rate of pellet velocity was independent of the initial velocity when the friction coefficient was the same value.

Alpha Particle Loss and Heat Load Assessment for Compact Stellarator Reactors*

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The ARIES Advanced System Study Group has embarked on a multi-year project to assess the potential of the compact stellarator concept as a fusion power plant. The initial phase of this study is to determine the optimum configuration (in terms of number of field periods and modular coils) that meets the physics performance goals and the engineering requirements that are crucial for an attractive reactor. Because of the non-axisymmetric nature of the magnetic geometry, it is possible that a significant number of the fusion-produced alpha particles will be lost from the plasma before complete thermalization. This loss of alpha particle power can affect the plasma power balance and the reactor operating point. More important are the effect of the exiting high-energy (MeV) particles on the design of divertors in terms of the heat load distribution, and the potential of blistering of the divertor coating by these particles.

In this paper, we report on an assessment of the alpha particle heat flux for a typical compact stellarator (CS) reactor configuration, as evaluated at the last closed magnetic surface (LCMS) of the plasma. Further work will be carried out to examine the resultant heat flux distribution in the plasma scrape-off layer in order to determine the optimum divertor design, and this will be reported at a later stage of the project. In the present analysis, we make use of the ORBIT3D particle orbit code [1] that follows the alpha particle guiding center from birth inside the plasma to its exit from the plasma. A CS configuration [2] with two field periods, $B_o = 5.5$ T, R = 7.5 m and A = 3.5, maintained by 16 coils at a beta value of 4%, is used as the reference equilibrium. In the standard case, an initial burst of lost alpha particles at the birth energy is observed with the subsequent lost energy spectrum moving towards lower values, resulting in an energy loss fraction of around 30%. The bulk of lost energetic alphas are concentrated in a toroidal strip just below the outboard mid-plane. The estimated peak heat load at the LCMS is well in excess of 10 MW/m^2, which can make the design of a divertor very challenging. Various methods to tailor the harmonic magnetic field components are being investigated to reduce the alpha heat loss, and the results will be presented.

* Work supported by the Office of Science, U.S. Department of Energy.

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The Correlation of ELM frequency on Pedestal Plasma Properties

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JET data for the correlations between pedestal plasma temperature and density with the frequency of type I ELMs are analyzed in the framework of a classical Focker-Planck approach. In first approximation we derive a functional dependence in steady state of ELMs assuming equality between the outward flux of plasma from the core and the loss of plasma from the pedestal into the SOL. In this assumption one expects equality between ELM frequency and Coulomb collisionality in the pedestal plasma. Our analysis of JET data shows that the derived formula for density dependence on temperature is obeyed in wide range of ELM size (including ELM III) but with varying coefficient that corresponds to the configurations of the engineering parameters. The values of this constant differ from that in the Focker-Planck solution for diffusion transport. Clear correlation of the constant with the input power is observed that indicates larger frequency of ELMs in comparison with the diffusion collisionality. It could be related to the impact of non-diffusion transport.

Preliminary estimations for the impact of plasma recycling and impurities on the quasi equilibrium state of pedestal plasma during ELM discharges are made. They will be verified with PIC simulation of the transport of plasma in the SOL and its interactions with the plates of the divertor .

Performance Characteristics of Actinide-Burning Fusion Power Plants*

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Burning actinides with fusion neutrons appears to be a promising option to destroy large quantity of actinides associated with spent fuel actinides and excess plutonium, and to utilize fertile fuel such as natural uranium to provide unlimited nuclear energy for mankind. To determine feasibility of transmutation with fusion neutrons, studies have been performed using a near-term fusion device as a neutron source. The actinides in the spent fuel are mainly Pu isotopes, although some minor actinides (MA), namely isotopes of Np, Am and Cm are also present. The excess plutonium material is mainly Pu239. Sub-criticality, reduction of high level waste (HLW) volume, and reduction of long-term risk of disposed HLW are among the major advantages offered by transmutation. The disadvantages of transmutation comparing to the once-through cycle scenario are concerns for plutonium proliferation and safety in handling enhanced radioactivity due to that associated with minor actinides, mainly Am and Cm isotopes, which may accumulate in large quantities during transmutation. Large fraction of minor actinides in the transmutation system appears also to be an important factor to reduce the performance and efficiency. Similar conclusions could be drawn for the fertile-burning nuclear energy generation fusion power plants.

Molten salt, which is a mixture of lithium fluoride and beryllium fluoride, is one of the transmutation blanket concepts studied recently. Molten salt can dissolve a small quantity of actinide salt in it and a fusion power plant based on the molten salt can become very attractive because of the possibility to minimize the actinide inventory in the transmutation plant. Furthermore, on-line removal of fission products and replenishing of destroyed actinides can be, in principle, operated in a molten salt fusion power plant.

In this paper, the focus of the presentation will be on two molten salt based fusion power plants. One of them is to burn spent fuel actinides, the other is to burn U238. Both power plants produce output energy larger than a fusion power plant would normally produce without including actinides. The spent fuel actinide burning power plant has two blanket options. One is a beryllium blanket to enhance the neutron multiplication and provide soft neutron spectrum for better fission burning of the actinides. The initial energy multiplication performance of this power plant can be as high as 180 when the criticality factor, k-eff, is 0.952. But after destruction of 3 actinide inventories in the blanket, the performance drops significantly, to an energy multiplication of 13 (k-eff 0.616). The other is a blanket without external beryllium. Due to a relatively harder neutron spectrum, modest performance is thus expected. But the performance is more stable along the burning time. The U238 burning fusion plant has much lower performance than in a spent fuel burning plant, because the replenishing material does not have a fissile component. Detailed performance of the various blanket concepts, its degradation during burn-up and the attendant evolution of isotopic concentrations of all actinides were obtained and will be discussed.

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The Production of ^{13}N using Beam-Target D-^3He Fusion

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The University of Wisconsin Inertial Electrostatic Confinement device has successfully been used to generate medical radioisotopes using the D-^3He fusion reaction. This project is unique in that it uses the by-products of fusion for a commercial application. Inertial Electrostatic Confinement (IEC) is a simple, relatively inexpensive method for sustaining steady-state fusion reactions. It may be developed into a small, semi-portable unit for producing isotopes, and perhaps a cheaper source of isotopes than what is currently available.

Electrostatic acceleration creates high-energy fusion ions to allow for the use of advanced reactions, such as D-^3He. The D-^3He reaction generates 14.7 MeV protons, and these protons have plenty of energy for the cross-sections needed in isotope production. Most of the D-^3He reactions in the IEC device have been found to occur as beam-target reactions in the cathode grid wires due to ion bombardment. This result led to the design of a beam-target method for generating isotopes.

The Wisconsin IEC project has focused initially on the production of isotopes used in Positron Emission Tomography (PET). The positron emitters used in PET scans all have relatively short half-lives. Their use would be greatly enhanced if a small or portable production unit were available. Short-lived species result in much lower residual radiation doses to patients. The isotope ^{13}N, with a ten-minute half-life, was created with the Wisconsin IEC device.

The cathode in the IEC device was replaced with a thin-walled, water-cooled, stainless steel tube. During operation, deuterium and helium-3 are driven into the tube wall, and further bombardment causes embedded fusion reactions to occur. Roughly half of the D-^3He protons travel deeper into the tube to reach the water flowing inside. The high-energy protons are able to generate ^{13}N through the $^{16}O(p,\alpha)^{13}N$ reaction using the oxygen in the water.

At a cathode voltage of 85 kV sustained for a few minutes, approximately 1.0 nCi of ^{13}N was created in a proof of principle experiment. The experimental setup and operation are described along with a discussion about ways to further increase the production yield. Although tens of mCi's are typically needed for PET medical scans, the experiments at the University of Wisconsin represent the first time D-^3He fusion has been used for this application.

Improvement of an Inertial Electrostatic Confinement Fusion Device by a Magnetron-Discharge-Based Built-In Ion Source

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An inertial-electrostatic confinement fusion (IECF) device is a compact discharge-type fusion neutron/proton source, consisting of a spherical vacuum chamber as an anode, and a central hollow cathode grid, where a glow discharge takes place. Positive ions generated are then accelerated to the center, and mostly after passing through the transparent hollow cathode, they are decelerated to stop in front of the anode, and again accelerated toward the central cathode again. These sloshing ions result in a high ion density at the center, and accordingly a high fusion reaction rate. Production of sufficient full-energy ions in the vicinity of the anode (chamber) and reduction of unnecessary charge-exchange (neutralization) with background gases are both essential to a drastic enhancement of ions' energy and lifetime leading to a higher fusion reaction rate.

For this purpose, we have proposed a magnetron discharge-type built-in ion source. Unlike other external ion sources, this ion source does not require additional positive extraction electric fields, which tend to make the ions to hit to the opposite anode. Moreover, the magnetron type built-in ion source has an extremely simple configuration to preserve the advantageous IECF features of compactness and robustness, essential to practical application.

In the previous experiments by use of a magnetron-type ion source consisting of a coaxial inner anode at a high voltage and a permanent magnet outside the grounded cathode, it is found that background gas pressure is reduced and ion beam energy increased, though this configuration is still not ideal in the viewpoint of ions' recirculation since produced ions have still considerable energy even at the opposite anode.

To resolve this demerit, we applied the inner coaxial electrode with a inner permanent magnet at a negative high voltage based on the numerical analyses. We designed a new magnetron-type ion source with a further refined configuration. It consists of an anode (grounded nozzle of 35 mm ϕ inner diameter), a coaxial cathode (20 mm ϕ diameter)at a negative voltage, and a cylindrical water-cooled permanent magnet (Nd-Fe) inside the cathode tube. Magnetron discharge then takes place by these localized electric and magnetic fields, and by shifting the inner cathode in the axial direction, evaluation of the influence of cathode position to discharge current and ion extraction characteristics is made.

This magnetron-type ion source shows a good performance, showing discharge current as high as 60 mA at a cathode voltage of 3.0 kV under H_2 gas pressure of 3.0 Pa. For further enhancement of discharge current, an optimal magnetic field distribution by use of additional magnets outside the nozzle is now being studied both experimentally and numerically.

Optimal Position of Ion Source for High Performance of IEC

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An inertial electrostatic confinement (IEC) fusion device is possibly used for the portable neutron sources. Recent paper reported that the neutrons of 1.8 x 10^8n/s are produced by the use of the glow discharge under the conditions of about 0.5 Pa deuterium gas pressure and the applied voltage of 180kV[1)]. In the most of IEC device, since gas pressure is too high, the accelerated ions lose their energy with the collisions with the neutral gas. The conditions of the high voltage and the low pressure are preferable because the energy which beam ions possess is used for the fusion reaction more efficiently. It, however, is difficult to keep the steady operation because the glow discharge at the low-pressure is not stable. One of the solutions is to equip ion sources such as a magnetron for supplying the ions near the anode.

We investigate the optimal position of ion sources to achieve a stable discharge even in the low pressure and stable discharge in the device by three-dimensional orbit following numerical code. This code includes atomic process, elastic collisions and the three dimensional electric field effects due to solid structures of the current feed-through and the cathode structures. We set D_2^+ ions at nearby inner side of the anode surface and trace the orbit of the ions until the ions are lost by the collisions with either the cathode or the feed-through or neutral gas.

The following results are obtained by the calculations.

1. The place where the ion source can realize the longest life of ion beam is 35 east longitudes and north latitude 11 degrees on the anode sphere (see Fig.1).
2. The optimal position equipping ion sources, in fact, depends on the operational gas pressure and the size of an ion source.
3. In order to expect that neutrons are generated effectively in low gas pressure by the beam-beam fusions, it is necessary to install two or more ion sources at the optimal positions on opposite sides.

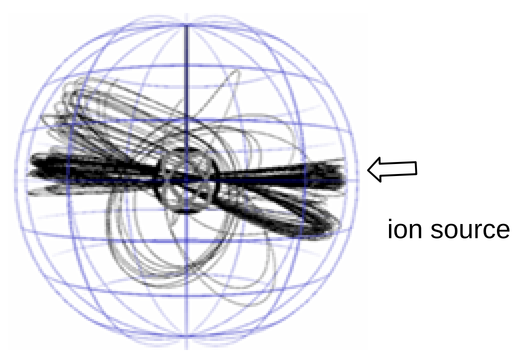

Fig.1 One example of ion beam orbit from the ion source

1) G. L. Kulcinski, et al. "6th US-Japan Workshop on Inertial Electrostatic Confinement Fusion" Tokyo Institute of Technology Yokohama, Japan Oct.20-21, 2003.

Numerical Study on Hollow Cathode Discharge of IEC Fusion

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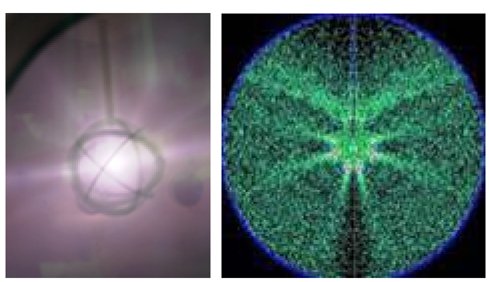
An inertial electrostatic confinement (IEC) fusion device is possibly used for the neutron sources that produce the neutrons of 1.8 x 10^8n/s by the use of the glow discharge under the conditions of about 0.5 Pa deuterium gas pressure and the applied voltage of 180kV[1)]. Since gas pressure in the most of IEC device, is so high that most nuclear fusion occurs between ion beam and neutral gas. The conditions of the high voltage and the low pressure are preferable, because the energy which beam ions possess is used for the fusion reaction more efficiently. It, however, is difficult to keep the robust operation because the glow discharge at the low-pressure is not stable.

We make the two-dimensional Monte Carlo PIC code in order to investigate the discharge characteristics in the IEC. The code includes 12 kinds of atomic processes and elastic collisions among electrons, D+ ions, D2+ ions and D3+ ions as well as the effects of the electric field deformed by the existence of the current feed through and the cathode structure. The code is initialed by the seed electrons existing between electrodes. The motions of each particle are traced by the Runge-Kutta method. During tracing the particles, atomic collisions are taken into account and new ions and electrons are generated by the collisions. The time behaviors of number of ion and electrons are observed. When the particles continue to increase, we identify that the glow discharge occurs.

We have the following results from the computations.

1. The relations between the gas pressure and the discharge voltage are consistent with those of the experiment.
2. The charged particle multiplication is proceeded dominantly by the ionizations of D_2 by fast D_2^+ ions and fast D_2 neutrals.
3. The fact that the discharge at a low pressure is unstable is explained by a less multiplication rate of ions
4. The ionization is rare in the region near the feed-through and its extension and lot of molecules are ionized near the hollow cathode.
5. We can reproduce the glow discharge pattern with several light spokes called “star mode” as shown in Fig. 1.



(a) (b)

Fig.1. “Star mode” (a) Experiment. (b) Simulation

1) G. L. Kulcinski, et al. “6th US-Japan Workshop on Inertial Electrostatic Confinement Fusion” Tokyo Institute of Technology Yokohama, Japan Oct.20-21, 2003.

Measurement of Ion Energy Distributions in a Cylindrical Inertial Electrostatic Confinement Fusion (C-IECF) Device

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An Inertial Electrostatic Confinement Fusion (IECF) is a device injecting ions and electrons towards the spherical center, trapping both species in the electrostatic self-field and giving rise to fusion reactions in the dense core. However, within the current experimental pressure (0.1Pa~10Pa), the majority of the fusion reaction occurs as collisions between the beam and the neutral gas because the gas density is much higher than density of high-energy particles, and the contribution of beam collision reaction (Beam-Beam Fusion) is little. The cylindrical IECF (C-IECF) device is proposed as simplification of the device at such an area where the increase of the density of the ion by spherical convergence is not important, and we have been carrying out experimental works aiming at search of the IECF principle using variety of the experiment shape and a handy nuclear fusion neutron source.

As the D-D fusion reaction cross-section increases with energy, it is clear that increase of the ion energy and of the ion longevity are shorter ways to increase neutron production and efficiency of the C-IECF device. Therefore, the method of using an external ion source to assisted glow discharge has been tried to reduce gas pressure, due to increase of ion average energy and decreasing of the ion loss by the charge exchange reaction between D_2^+ beam and background gas. As results, operating gas pressure has reduced from 1.5 Pa to 0.5 Pa, the neutron production has increased about 20%.

These results are considered as shown in the above-mentioned scenario. To confirm that an increase in the ion energy is a factor of an increase in the neutron production, the ion energy distribution of the C-IECF device has been measured. As the method, the part of the neutral particle beam (with the ion energy distribution and strong relation) was drawn out from the hole made at center of the anode, and was gone through re-ionization chamber, then it’s energy was measured by energy analyzer. The primary result shows the change of the ion energy distribution at the low energy element along the pressure.

Fast Neutral Generation by Charge Exchange Reaction and its Effect on Nuclear Burning in Inertial Electrostatic Confinement Fusion Systems

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The inertial electrostatic confinement (IEC) is a concept for electrostatically confining high-energy fuel ions in potential well. In ideal IEC plasmas, the ions converge toward the center of the device, and their space charge forms a virtual anode. The IEC fusion system has intrinsic potential for earlier practical use of fusion energy as a compact and economical neutron/proton source. So far, neutrons/protons more than $10^8 ns^{-1}$ produced by D(d,n)^{3}He(^{3}He(d,p) ^{4}He) fusion reactions have been observed on several devices. To improve further the device performance, it is important to understand the physics of the IEC plasmas.

In the device several kinds of ion-neutral collisions occur, e.g. elastic-scattering, ionization, charge exchange in parallel with the fusion reaction. Especially, the charge exchange reaction (CX) changes accelerated ions to fast neutrals which can cause fusion reaction with background neutral gas. In previous researches, it has been shown that fusion reaction between the fast neutral and background gas is comparable with those between ion and background gas [1, 2]. We have previously examined correlation between the ion distribution function and neutron production rate between ion and background gas in spherical IEC devices [3]. The fusion reaction between fast neutral, which produced by CX, and background gas would also be influenced by the shape of the ion distribution function. In this paper, we investigate the influence of the ion distribution function on fusion reaction between fast neutral and background gas.

We assume the spherical IEC device. Within the spherical cathode, the potential structure is determined by solving the Poisson-Equation, and the Child-Langmuir radial potential is assumed outside the spherical cathode region. The Boltzmann-Equation for first neutrals is solved considering the fast neutral generation and loss by CX reactions, together with the particle transport loss from the device. It is shown that fusion reactions carried by the CX fast neutrals becomes appreciable especially in the high-voltage operations, and the fusion reaction between fast neutral and background gas is sensitively affected by the shape of the ion distribution function.

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One Dimensional Simulation of an Inertial Electrostatic Confinement Fusion at Low Gas Pressure Operation

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In order to find conditions which make it possible to obtain large output/input ratio in a Spherical Inertial Electrostatic Confinement Fusion (S-IECF), we have developed a one-dimensional simulation code which can be directly compared with experimental results, and have analyzed characteristics. In this study, we have examined "assisted glow discharge" in which ions are supplied by additional ion source to maintain discharge at low gas pressure.

We have developed a 1-D Particle in Cell code with Monte Carlo collision scheme. To simulate the IECF device, several special methods are required, (a) modeling of the cathode grid which has high transparency, (b) implementation of atomic and molecular processes to simulate deuterium gas operation, and (c) calculation of spatial distribution and amount of fusion reaction. Using this code, potential distribution, behavior of each particles and neutron production rate (NPR) can be calculated.

To simulate "assisted glow discharge", calculation injecting D_2^+ from the position of the anode has been done. At low gas pressure (<0.5Pa), the discharge can not be sustained for lack of charged particle without ion source, and adding it enable self-maintaining discharge. Gas pressure (P), applied voltage (V) and discharge current (I) can be controlled independently by adjusting the ion current injected from the ion source (I_{assist}). Dependency on the P of the I and the NPR are calculated with constant I_{assist} and V_c. Simulation results show that at low gas pressure, (a) the frequency of collision between the beam particles and the background gas is reduced (b) energy loss of each ion is decreased, though the NPR does not increase because number of particle decrease. Comparing contribution to the NPR of each species and number of each species, it is found that D^+ is more reactive than D_2^+. Therefore when changing injected species to D^+, larger NPR is obtained. At low gas pressure, simulation result that most part of discharge current is occupied by secondary electron current (e.g. 80% at P = 0.2mTorr and V_c = 60kV) suggests that reduction of secondary electron emission coefficient is effective to improve the ratio of the output (NPR) to the input power, which is important to apply the IECF to practical use.

From these results, we consider that low gas pressure operation of IECF by adding external ion source is effective to improve the efficiency.

Influence of the Electrode Spacing on the Performance Characteristics of Inertial Electrostatic Confinement Fusion in Low Pressure Operation

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To understand the phenomenon in an inertial electrostatic confinement fusion (IECF) device, we have been developing an 1-D particle simulation code including the atomic processes between the ion beam, the neutral beam, and background gases. Now, the characteristics of electrical discharge (voltage dependence on gas pressure et al.) and of the neutron generation in the experiments can be almost reproduced by the simulation. Moreover, it has been quantitatively clarified through these simulations that the loop of the neutralization by charge-exchange and re-ionization becomes predominant in the self-sustaining mechanism of IECF discharge when gas pressure is deceased less than 1 Pa. This result indicates that the mean-free-path of the reionization reaction influences the relation of the anode-cathode spacing and discharge pressure. On the other hand, it is understood from the calculation of the space distribution of the nuclear fusion reaction that the nuclear fusion reaction between energetic ions and background gas has the peak value near the cathode where electric field strength is strong.

In the on-going IECF experiments, further reduction of the operation pressure using external plasma sources is actively investigated to realize beam current accumulation by the ion reciprocation between electrodes which is one of key features of the IECF concept. In this case, the operation of IECF is no more self-sustained discharge mode, and the restriction of electrode spacing may be disappeared. Therefore, it is thought that increase of the electric field strength by shortening the distance between anode and cathode can promptly accelerate the ion, and the loss by the charge exchange in low energy can be decreased.

In this presentation, we examine the effects of the electrode spacing in the low gas pressure region (<0.1 Pa) on IECF operation characteristics and fusion reaction rate by the 1-D particle simulation.

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Pulsed Operation of a Compact Fusion Neutron Source Using a High-Voltage Pulse Generator Developed for Landmine Detection

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Preliminary experimental results of pulsed neutron source based on discharge-type beam fusion called Inertial Electrostatic Confinement Fusion (IECF) for landmine detection are presented. In Japan, a research and development project for constructing the advanced anti-personnel landmine detection system by using IECF, which is effective not only for metal landmines but also for plastic ones, is now in progress. This project consists of some R&D topics, and one of them is R&D of a high-voltage pulse generator system specialized for landmine detection, which can be used in the severe environment such as that in Afghanistan. Thus the circuit configuration of pulse generator system was discussed, and a prototype of the system for landmine detection was designed and made in consideration of compactness, lightness, cooling performance and robustness. By using this prototype system, a conventional IECF device was operated as a preliminary experiment. As a result, it was confirmed that the proposed circuit configuration is suitable for landmine detection system, and the results follow the empirical law obtained by the previous experiments. The maximum neutron production rate of 2.0×10^8 n/s was obtained at a pulsed discharge of –51 kV, 7.3 A. With this result, the initial target neutron production rate of the project, 10^8 n/s, has been achieved and the system is found to be very promising because the performance of this prototype system is a half of that of the planned final system. Therefore, at the target pulsed discharge of –90 kV, 10 A, a neutron production rate of 10^9 n/s, which leads to the reduction of the detection time and the improvement of the identification accuracy, is expected to be achieved finally.

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Implantation of D^+ and He^+ in Candidate Fusion First Wall Materials

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The effects of high temperature (700-1200 ºC) implantation of deuterium and helium in candidate fusion first wall materials were studied in the University of Wisconsin Inertial Electrostatic Confinement (IEC) device. Tungsten "foam", single crystal tungsten, and a W-25%Re alloy were compared to previous tungsten powder metallurgy samples studied in the IEC device for the High Average Power Laser (HAPL) program. Scanning electron microscopy was performed to evaluate changes in surface morphology for various ion fluences at temperature ranges comparable to first wall temperatures. Preliminary results show that no deformations occur with deuterium implantation up to $2x10^{18}$ D^+/cm^2 at 1200 ºC polycrystalline tungsten samples. However, helium fluences in excess of $4x10^{17}$ He^+/cm^2 show extensive pore formation at 800 ºC. These changes will have an impact on the lifetime of thin tungsten coatings on the first walls of inertial and magnetic confinement fusion reactors.

Optimizing Neutron Production Rates from D-D Fusion in an Inertial Electrostatic Confinement Device

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Detection of explosives has been identified as a near term commercial opportunity using a fusion plasma. Typical explosive compositions contain low Z material (C, N, O) which are not easily detected using conventional x-rays or metal detectors. However, 2.45 MeV neutrons produced in a D-D fusion reaction can be used for explosives detection in suitcases, packages, shipping containers, or other clandestine materials.

Steady-state D-D operation is possible using Inertial Electrostatic Confinement (IEC) fusion. The University of Wisconsin IEC device has produced D-D neutrons at 1.8 x 10^8 neutrons/second at a cathode voltage of 166 kV and a meter current of 68 mA. These neutron production rates are approaching the levels required for the detection of explosives. In order to increase and optimize the neutron production rate in the IEC device, experiments were performed to determine the affect on the neutron production rate by altering the cathode's size (diameter), geometry, and material composition. Preliminary results indicate that significant differences in neutron production rates are not achieved by altering the geometry or material composition of the cathode. However, the neutron production rate was found to increase approximately 20% by doubling the cathode's diameter from 10 cm to 20 cm. In addition, increasing the cathode voltage from 34 kV to 94 kV at a meter current of 30 mA increased the neutron production rate from 1.24 x 10^6 n/s to 2.83 x 10^7 n/s.

D-^{3}He Proton Energy Distribution from an IEC Device*

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High energy protons are released from the D(^{3}He, p)^{4}He reactions in an IEC device. However, parasitic D(d,p)T and D(d,n)^{3}He reactions also occur simultaneously with the D(^{3}He, p)^{4}He reactions. The proton energy spectrum from the DD and D-^{3}He reactions have peaks at 3.02 MeV and 14.7 MeV respectively. These peaks show sizeable broadening caused by both Doppler broadening and energy staggering in the 700 μm thick Si detector. There is also a high-energy tail structure associated with the D-^{3}He proton peak that extends up to 10 MeV, which is the maximum proton energy that can be deposited in this detector. The Bragg response (stopping power variation with energy) of the Si detector has been identified as the cause of both the 14.7 MeV protons showing up at ~5 MeV and the associated high energy tail. The present paper calculates the proton energy spectrum using the Monte Carlo Stopping and Range of Ions in Matter (SRIM) code. Good agreement (within ~10%) between the experimental observations and theoretical predictions has been obtained.

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Design of an Ion Source for Low Pressure IEC Operation*

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Inertial Electrostatic Confinement (IEC) systems have been used for decades to investigate the feasibility of fusion, including the use of advanced fuels, such as D-D and D-^{3}He. These reactions require high center-of-mass energies (50-100 keV). Typically, fusion fuels are ionized to provide the source of ions inside of the device, restricting operating pressures to ~1mTorr or greater. In this pressure range, ion-neutral collisions and charge exchange occur frequently and dominate the ion dynamics. In order to reduce the negative effects of atomic processes and increase center-of-mass collision energies, the IEC must be operated at much lower pressures, < 50μTorr, where ion-neutral collisions will be infrequent. To this end, an ion injection system has been constructed, which consists of a high density helicon plasma generator and a high-voltage, differentially pumped extraction system. Plasma currents of up to 10 mA of ^{4}He have been achieved into a vacuum of 50μTorr. The design of the hardware, recent results, and the capability of this source to generate a beam of ionized ^{3}He for IEC fusion will be discussed.

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An Electrostatic Confinement Experiment to Explore the Periodically Oscillating Plasma Sphere

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Theoretical work[1,2] has suggested that a tiny oscillating ion cloud (Periodically Oscillating Plasma Sphere or POPS) may undergo a self-similar collapse that can result in the periodic and simultaneous attainment ultra-high densities and temperatures. Theoretical projections[1] indicate that such a system may have net fusion gain even for an advanced fuel such as Deuterium-Deuterium. Schemes have also been suggested where a massively modular system consisting of tens of thousands of these spheres can lead to a very high mass power density device (similar to a fission light water reactor)[1] while still having conventional wall loads (~ 1 MW/m^2).

POPS also has the feature that the total fusion power scales inversely with the device radius.[1] This favorable surface to volume scaling leads to high mass power density and thus a device which will be competitive with fission, coal, gas and renewable energy sources. The lightweight, modular design is also an excellent fit for the space propulsion systems which will be required for interplanetary space travel.

The fusion energy development path for the POPS system is also very different from conventional fusion systems. Conventional systems tend to get larger and more expensive with each generation. (For instance, the next generation magnetic confinement device (ITER) is projected to cost twelve billion dollars.) For POPS, we will demonstrate the physics on a single cell so each generation device should be smaller than the previous generation. The costs of any successive device should be similar to our present device that was less than $100k.

Thus, this device combines the following three attributes: a high mass power density device with favorable economics, a low cost development path where no device costs more than $100k, and a system which can operate with advanced fuels. However, a number of issues need to be addressed to determine the efficacy of the POPS scheme. Some of these issues are: phase locking and control of the ion oscillations, space charge neutralization during the ion collapse phase, electron cloud uniformity and control, stability of the electron cloud, and maintenance of the virtual cathode. These issues are being addressed both theoretically and experimentally.

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Initial Activation Assessment for ARIES Compact Stellarator Power Plant

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The ARIES team is moving forward with a new study: the ARIES-CS compact stellarator power plant. It is a pioneer design that combines advanced physics and engineering approaches to minimize the major radius and hence the overall size of the machine. Regarding the engineering activities, five blanket/shield systems have been proposed (Flibe/FS/Be, LiPb/SiC, LiPb/FS/He, Li/FS/He, and Li_4SiO_4/FS/Be/He) and an effort is underway to integrate the internals (blanket, shield, and vacuum vessel) and develop a credible maintenance scheme that supports high availability of 85% or more. The blanket covers the majority of the first wall (~90%). At a few localized areas where the magnet moves closer to the plasma, a highly efficient, thin WC-based shield replaces the blanket to protect the magnet against radiation. We developed two radial builds for each blanket concept: one for the nominal blanket, shield, and vacuum vessel region and the other for the WC-shield and vacuum vessel region.

As inputs for the safety assessment that frequently requires knowledge of the activation parameters, we estimated the highest possible activity, decay heat, and waste disposal rating on the time scale after shutdown for the two radial builds mentioned above. In the absence of a reference design, we selected for this study two widely different systems (LiPb/SiC and LiPb/FS/He) employing SiC/SiC composites and low-activation ferritic steel (FS) as structural materials. The ALARA activation code, DANTSYS transport code, and FENDL-2 data have been used throughout the study. Key design parameters for this initial assessment include an average neutron wall loading of 2 MW/m^2, a first wall lifetime of 5-6 FPY, and a permanent component lifetime of 40 FPY. It is found that the WC-shield generates the highest activity and decay heat among all components. The SiC structure of the blanket generates the lowest activity and decay heat and their initial values drop rapidly by 3-4 orders of magnitude at one day after shutdown. An estimate for the peak temperature at the SiC and FS structures during an accident is currently underway making use of the decay heat as a heat source. If the temperature exceeds the limit of the reusability of the structure, a separate decay heat removal loop will be installed within the WC-shield in particular to control its temperature during an accident.

Among the three radwaste management approaches envisioned for ARIES-CS, the disposal and clearance options have been investigated in detail. This paper focuses on the disposal issue while a companion paper covers the details of the clearance approach[1]. We evaluated the waste disposal rating (WDR) for a compacted waste using the most conservative waste disposal limits developed by Fetter and NRC-10CFR61. Our results show that all components including the WC-shield qualify as Class C low-level waste (LLW) at the end of a 100 y storage period following the decommissioning of the plant. Moreover, the SiC blanket, vacuum vessel, and magnet offer very low WDR (< 0.1) to the extent that a Class A LLW seems achievable for these components. On this last point, we will discuss the split between the Class A and Class C wastes, emphasizing our motivation to lower the level and minimize the volume of ARIES-CS radwaste.

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Compressed Gas Safety for Experimental Fusion Facilities

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Experimental fusion facilities present a variety of hazards to the operators and staff. There are unique or specialized hazards, including magnetic fields, cryogens, radio frequency emissions, and vacuum reservoirs. There are also more general industrial hazards, such as a wide variety of electrical power distributed throughout the facility, instrument air, cooling water, crane and hoist loads, working at height, and handling compressed gas cylinders. This paper outlines the hazards associated with compressed gas cylinders and methods of treatment to provide for compressed gas safety. This information should be of interest to personnel at both magnetic and inertial fusion experiments.

A variety of gases are used in fusion research. Nitrogen is used as a fill gas in vacuum vessels and in gloveboxes to reduce intrusion of humid room air, hydrogen and deuterium gases are used as fuel sources, and P-10 gas (10% by volume methane and 90% argon) is used for a cover gas in radiation counters such as ionization chambers and portal-type contamination monitors. Sometimes halon gas or carbon dioxide gas is used for fire suppression. Other gases may be used to cool diagnostic devices or as lasing gases for laser diagnostics, including nitrogen, carbon dioxide, neon, argon, and others. A typical 140 cm tall by 23 cm diameter gas cylinder from a gas supply company is pressurized to ~13 MPa (2,000 psia) and generally weighs about 59 kg (130 pounds). In the press of experiment activity, proper treatment of gas cylinders can be overlooked.

The US Department of Energy (DOE) Occurrence Reporting and Processing System off-normal events database has been searched to identify incidents from 1990 and on, across the DOE complex, that are pertinent to compressed gas safety in fusion research. The findings are summarized in this paper; they include system parts such as metal parts and gauges that have been expelled under gas pressure, incorrect gases being plumbed to an experiment, cylinder isolation valve leaks, gas cylinder "missiles", cylinder through-wall 'pinhole' leaks, and other events. Preliminary estimation of the frequency range of several of these types of events is given; the frequencies are low but these are nonetheless credible events. Thus far, no US fusion experiment facilities have reported any gas cylinder events to the DOE. However, in some DOE-funded facilities the treatment and handling of these cylinders is not safety conservative. Other safety-related events with compressed gas cylinders are also summarized in this paper, including US Nuclear Regulatory Commission event descriptions and a few chemical process industry case histories.

A calculation of the thrust force available from gas exiting the bore of a sheared valve, assuming a typical gas cylinder, has been performed and results are given for a variety of gases that are often used at fusion facilities. These calculations, and the historical event accounts, show that there is ample force available to penetrate sheet metal and cinder block walls, gouge concrete floors, or to send a cylinder aloft. Treating gas cylinders with respect and keeping cylinders well restrained while in use is a prudent course of action for all fusion researchers.

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Activities of the US-Japan Safety Monitor Joint Working Group

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This paper documents the activities of the US-Japan exchange in the area of personnel safety at magnetic and laser fusion experiments. This exchange is part of the US-Japan Bilateral Agreement on Fusion Research. The US and Japan have hosted visiting researchers at each of the large tokamaks, major experiments and fusion research centers for decades. Most of these exchanges have been performed without any safety incidents in either country. Unfortunately, in early 1992, there was a near-miss event of high safety concern in the US.

The near-miss industrial accident with a visiting Japanese scientist to the US was the impetus for forming the Joint Working Group (JWG) on Fusion Safety under the Bilateral Agreement. This Fusion Safety exchange has been under way for over ten years and has been successful in providing many safety insights for both US and Japanese facility personnel at national institutes and universities. The US members of the JWG are funded under the auspices of the US Department of Energy, Office of Fusion Energy Science. The background, reasons and activities of the JWG are described in this paper, including a list of the participants from both countries, a list of the facilities that have been visited for safety walkthroughs, and the main safety issues examined during the brief JWG walkthrough visits. Major facilities in both countries have been visited, including the DIII-D tokamak and the Tokamak Fusion Test Reactor in the US; and the JT-60U tokamak, the Large Helical Device stellarator, the GAMMA-10 mirror machine, and the GEKKO-XII laser fusion experiment in Japan. Smaller scale university experiments have also been visited in both countries.

Based on these visits and the operating experiences of experiments at the JWG members' home institutions, the JWG members have compiled some best practices and formulated several ideas to enhance safety at fusion experiments. These operational safety practices and ideas are briefly discussed in the paper. These practices and ideas include supplementing the written safety signs with internationally recognized pictogram signs, using daily safety checklists before commencing experiment operation, performing a visual search and sweep before commencing operation, appointing a key person of the day to track safety and operations issues, holding brief pre-operation meetings, and instituting once-a-month cleanup days. All of these practices and ideas help to promote safe operation of any fusion experiment, large or small, either magnetic or inertial.

The near-term future plans of the Safety Monitor JWG are also discussed. The JWG plans to continue with safety walkthroughs at the present frequency of one walkthrough team visiting the other country every second year to keep safety a prominent part of the operation of fusion experiments. The US JWG members completed a safety walkthrough of Japanese fusion experiments in February 2004. The next scheduled visit is for Japanese JWG safety personnel to visit US facilities in late 2005 or early 2006.

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The Worker Exposure Failure Modes and Effects Analysis

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This paper presents a new approach to quantitative evaluation of worker risks from possible failures of co-located equipment in the complex environment of a magnetic or inertial confinement fusion experiment. For next-step experiments such as the International Thermonuclear Experimental Reactor (ITER) or the National Ignition Facility (NIF), the systems and equipment will be larger, handle more throughput or power, and will, in general, be more robust than past experiments. Despite the large size of these experiments, there is congestion as all the required systems and equipment share rooms so that they can be plumbed or connected to the tokamak or laser chamber. These systems and equipment are necessary to operate the machine, but the complicated equipment rooms and access pathways for piping runs or cable chases also pose a new, higher level of hazard for workers who will perform the necessary hands-on maintenance tasks.

Conventional safety assessment methods, such as the well-known Job Hazards Analysis (JHA) endorsed by the US Occupational Safety and Health Administration (OSHA), focus on the task at hand; these methods define the hazards inherent with the steps of that task and identify methods that can be used to mitigate those hazards. Presuming that conventional worker safety assessment methods will provide adequate protection for the task at hand, completing JHA or similar analyses will still leave the worker potentially vulnerable to hazards from co-located equipment and systems in these complex facilities. The new analysis approach, called the Worker Exposure Failure Modes and Effects Analysis (WE-FMEA) is a method to analyze the nearby equipment and the work environment for equipment failure and other inherent hazards, and then develop scenarios of workers' exposure to the hazards. The proven FMEA technique lends its structure to provide a systematic, thorough treatment to identify the potential hazards to the workers. Only the equipment failure modes that can threaten the workers with exposure to an energy source or a hazardous material are addressed in this analysis. Once identified, the exposure scenarios can be evaluated for the severity of the worker hazards and quantitative worker risk values can be calculated.

The US DOE Fusion Safety Standard directs that fusion facility workers shall be protected from routine industrial hazards to a level commensurate with that of comparable industrial facilities, and that the US OSHA regulations will be followed to provide worker safety. Therefore, data from comparable industrial activities that are similarly regulated can be used as a comparison point to the WE-FMEA findings. Each quantified worker risk scenario identified in the WE-FMEA can be compared to existing statistical data on worker injuries from a comparable industry; WE-FMEA scenarios that are shown to be unacceptably high risk can be analyzed in more detail to agree on the proper means to reduce or mitigate the hazard. In this paper, the WE-FMEA approach is described, its strengths and potential weaknesses are discussed, and a cooling system maintenance example is given. This new approach can support worker safety assessment in ITER, NIF, and other next-step experiments.

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Design Constraints for Liquid-Protected Divertors

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Work on plasma surface interactions for liquid-surface-protected plasma facing components has been performed by the Advanced Limiter-Divertor Plasma-facing Systems (ALPS) Program. Fusion reactor designs with liquid first walls and divertors have also been investigated as a part of the APEX program. Operating temperature windows were established based on the fluid properties and power conversion efficiency requirements. Constraints were also imposed on the maximum allowable fluid surface temperature in order to limit the primary source of plasma impurities, *viz*. vaporization.

In this study, an additional constraint for the maximum allowable wall and coolant surface temperature gradients (*i.e.*, spatial heat flux gradients) has been quantified. A mechanistic transient three-dimensional model using the level contour reconstruction method has been used to follow the evolution of the liquid film free surface above a non-isothermal solid surface. Spatial variations in the wall and liquid surface temperatures are expected due to variations in the wall loading; thermocapillary forces created by such temperature gradients can lead to film rupture and dry spot formation in regions of elevated local temperatures. Parametric studies are performed for various coolant properties, liquid film thicknesses, mean coolant temperatures, and spatial temperature gradients. The results are used to develop generalized charts for the maximum allowable spatial temperature gradients for different coolants over a wide range of operating conditions. These charts will allow reactor designers to identify design windows for successful operation of liquid-protected plasma-facing components.

Particle Control by Lithium-Gettered Moving-Surface Plasma-Facing Components in Steady State Magnetic Fusion Devices

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Ever since the discovery of TFTR's Supershot in late 80's, high performance confinement plasmas have often been found to favor low edge recycling conditions. Therefore, wall conditioning such as boronization has routinely been conducted in many confinement devices. Unfortunately, however, due to the surface saturation with implanted particles, the efficacy of boronization to maintain particle recycling at low levels has finite lifetime, necessitating re-conditioning. This clearly points to the need for enabling wall concept development to provide reduced recycling even at steady state for the future long-pulse or steady state fusion devices.

As a possible resolution to this steady state plasma-surface interactions issue, the concept of moving-surface plasma-facing component (MS-PFC) was proposed while ago [1]. Recently, laboratory-scale proof-of-principle (PoP) experiments have successfully been conducted, employing a continuously Ti-gettered rotating drum exposed to steady state hydrogen plasmas, and the results indicate that wall recycling reduced down to 95% can be maintained at steady state [2]. In our previous paper [3], these experiments were extended, using Li as the getter, and some of the preliminary but encouraging results, including steady state recycling of 90%, were reported.

In the present work, the particle control capability of the Li-gettered rotating drum has been extensively investigated with the particular emphasis on understanding the relationship between the deposition rate and steady state recycling level. Shown in Fig. 2 are the recycling data taken from a continuously Li-gettered rotating drum at the deposition rate of 13 /s. Notice that a 4% reduction is attained at steady state in hydrogen recycling, measured with H_α spectroscopy, over the Li-gettered rotating drum. To conduct particle balance analysis, similar to that performed for the Ti-gettered case [2], sticking coefficient measurements have also been conducted for Li. Results indicate that the sticking coefficients of hydrogen molecules and hydrogenic species in hydrogen plasmas are 7.5 x 10^{-4} and 0.31, respectively. Detailed analysis is under way.

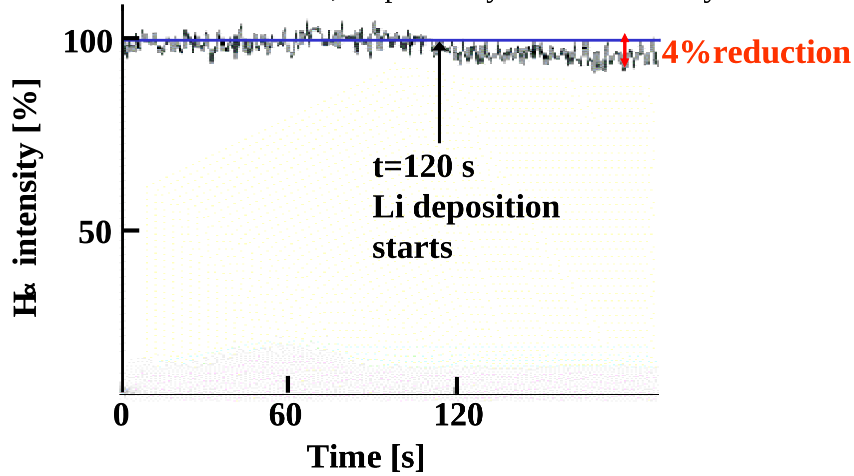

Fig. 1 Reduced recycling at steady state demonstrated by the Li-gettered rotating drum.

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Optimization of the HETS He-cooled divertor concept: Thermal-Fluid and structural analysis

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The High Efficiency Thermal Shield (HETS) concept was proposed by ENEA for divertor application in the context of the ITER project and as part of the European Power Plant Conceptual Study. The design is modular, and the segment dimensions are of the order of a few centimeters for the purpose of limiting the induced mechanical and thermal stresses. The requirement for high temperature operation led to the structural material choices of tungsten for the armour, and tungsten alloy for the pressure-retaining boundary.

Tungsten and alloys have an operating temperature range 300-1300^{o}C, limited by the ductile to brittle transition (DBT) and recrystallisation temperatures respectively. The DBT temperature under neutron irradiation is estimated to be around 600^{o}C, which narrows the “design window” to a range of 600-1300^{o}C.

The coolant is helium gas operated at pressures in the range 10-14 MPa with inlet temperatures in the range of 500-800^{o}C. The final choice of operating parameters is made in conjunction with the choice of mass flow rate, in order to keep temperature values within the structural material operating limits, and pressure drop and pumping power to a minimum.

The geometrical complexity of the designs made prediction of heat transfer coefficients, needed for conducting thermal and structural analysis difficult, and the calculated values from empirical correlations uncertain.

Previous studies demonstrated that the design assumptions were conservative. This paper presents and summarizes results of thermal-fluid and structural analyses, with different heat flux loads, fluid pressures and inlet velocities. The inlet coolant velocity was varied from 175 to 300m/s, resulting in mass flow rate variations in the range 0.052-0.0891kg/s, and the incident heat flux was raised from 10MW/m^2 to 15-20MW/m^2. Two helium gas pressures were studied: 10MPa and 14MPa, to see the effect of pressure on the resulting primary stress and strain.

The computational fluid dynamics analysis demonstrates that the flow passage with a sharp corner at the point of flow reversal behaves like an abrupt enlargement, leading to considerable pressure losses as compared to the results obtained by rounding the corner. Rounding the sharp corner causes the passage to behave like a diffuser, where pressure is recovering due to the flow cross sectional area expansion, leading to reduced total pressure losses, without any degradation of the thermal performance of the component.

The finite element structural analyses results demonstrate that the static stress requirements, including the limits for stress concentration regions, are met or are within the calculation uncertainties. Also, the deformation limits are satisfied, as the strains calculated are within the 1% average and 5% local values specified in the design code, although an inelastic analysis might have produced higher values than the 0.123% calculated. A lifetime evaluation was not performed as time dependent data for tungsten alloys, i.e. creep (thermal and irradiation) and fatigue, are scarce and not code qualified.

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Experimental Analysis of Soaker Hose Concept for First Wall/Divertor Application

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To allow the first wall/divertor to remove the intense radiative surface flux and to prevent it from being damaged by, creating a liquid metal free surface wall is one of the constructive goals in the fusion engineering field. Several concepts have been considered including film flow and free surface jets. However, these ideas may not be feasible ways to flow liquid metal crossing a magnetic field. The soaker hose idea was proposed to this first wall/divertor application by flowing liquid metal along the field lines or flowing toroidally. In such a concept, the fluid is introduced into the first wall by the differential pressure head, and current is applied to the liquid to create a Lorentz force to redirect the flow. Although numerical analyses have been performed for feasibility evaluations, few experiments have been carried out. The objective of this work is to analyze the effectiveness of the soaker hose concept experimentally.

In the present experiments, current added to the fluid metal can be up to 30A, while the magnetic field, which is produced by placing the two permanent magnets facing each other, is approximately 0.20T. Acrylic plastic, an electrically isolated material, is used as the hose material, and Ga-In-Sn alloy, fluid metal, is introduced into the magnetic field by static pressure head. Important parameters in this experiment are inlet velocity, the strength of the Lorentz force (which is the cross product of the magnetic field and the current density) and the fluid flow path.

Preliminary experimental results, from both vertical and horizontal hose setups, have shown clearly that the Lorentz force has acted to the liquid metal if current is added perpendicularly to the magnetic field; as seen in the previous numerical analysis. Specifically, the MHD Lorentz force has modified the shape of the flow surface and retarded the inlet flow. This retarding force has resulted in decreasing inlet velocity as the supplied current increases, or a 5% decrease in inlet velocity at a supplied current magnitude of 30 A. However, to allow the fluid to completely cover the first wall/divertor area, a stronger Lorentz force is needed to overcome the gravitational force; as already shown in the vertical setup where the fluid has been pulled together quickly by gravity rather than been pushed toroidally by MHD forces. On the contrary, in the horizontal setting, the same Lorentz force flattens the flow and directs the flow more effectively. In both settings, however, the surface tension and surface wetting play a significant role in determining flow characteristics. These are challenging problems that need to be resolved in order to put forward this concept for practical application.

Pulsed X-Ray Exposures and Modeling for Tungsten as an IFE First Wall Material

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Dry-wall inertial fusion energy (IFE) power plants must survive repeated exposure to target threats that include x-rays, ions, and neutrons. While this exposure may lead to sputtering, exfoliation, transmutation, and swelling, more basic effects are thermomechanical in nature. In the present work, we use the newly developed RadHeat code to predict time-temperature profiles in a tungsten armor, which has been proposed for use in an IFE power plant.

The XAPPER x-ray damage experiment can be used to simulate thermal effects by operating at an x-ray fluence that produces similar peak temperatures, temperature gradients, or thermomechanical stresses. The x-ray source used within the XAPPER facility was designed and built by PLEX LLC. It produces short (30-50 ns) pulses of soft (80-150 eV) x-rays and focuses them upon a sample. X-ray fluences in excess of 1 J/cm^2 are possible. XAPPER can operate at up to 10 Hz for tens of millions of pulses before requiring minor maintenance.

Using RadHeat, we determine the XAPPER x-ray fluence needed to simulate the thermomechanical effects expected in a typical IFE case of interest. We have exposed a set of tungsten samples to varying fluences and numbers of pulses. Here, we report our findings and detail directions for future experiments and modeling.

DIONISOS: A new experiment on the dynamics of plasma-surface interactions

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The plasma-surface interface composes one of the most interesting and important fields of study in plasma physics, and in the application of plasma physics to magnetic confinement fusion research. Both the plasma and the material surface simultaneously change characteristics due to their interaction, and affect each other in a complex, non-linear manner. This dynamic interaction leads to several effects, such as net erosion of plasma-facing surfaces, tritium retention in deposit layers and plasma impurity contamination, which are all estimated to limit the viability of long-pulse plasma fusion devices. However, in general, these phenomena remain relatively poorly understood, primarily due to the lack of proper plasma-surface diagnosis.

A new experimental facility DIONISOS (Dynamics of ION Sputtering and Implantation On Surfaces) is described. The innovative feature of the facility is its ability to measure accurately the real-time response of the material surface to the plasma bombardment using in situ high-energy ion beam analysis. The facility couples a dual-source 1.7 MV tandem ion accelerator to a plasma exposure chamber. A high current (~100 μA) sputtering negative ion source on the accelerator allows for in-situ implantation of surface and for the simulation of energetic particle damage found in a fusion reactor environment. A low current (~ 1 μA) RF ion source provides He ion beams for in-situ ion beam surface analysis (IBA). The IBA measures depth-resolved elemental composition, including hydrogen isotopes, in the first few μm of the surface, which is the region of interest for plasma-surface interactions. The facility incorporates flexible plasma/beam exposure geometry to provide spatially and temporally resolved measurements of net erosion rates, film growth and hydrogen trapping. The IBA is complemented by local plasma optical emission diagnostics that relate the material efflux to the net erosion.

Initial operation will use a helicon plasma source to provide steady-state plasma exposure over a wide range of plasma densities. The use of pulsed plasma guns are also planned to study transient effects on the material surface. Active heating and cooling of the sample will control temperature during plasma exposures.

We will briefly describe specific research topics

- The effect of ionization mean-free path on net erosion rates.
- The dynamics of hydrogenic/tritium fuel trapping in plasma-deposited films, for single or multiple species materials.
- Mixing and erosion of tungsten / carbon plasma-facing surfaces.
- The dynamic release of fuel and impurity particles from surfaces exposed to transient, high-density plasmas.

The Flowing Liquid-Metal Retention Experiment (FLIRE) Results

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The flowing liquid retention experiment (FLIRE) at the University of Illinois measures the retention of implanted helium and hydrogen in a flowing liquid stream. The ability of flowing liquid metal surfaces to withstand high heat fluxes without permanent damage makes them attractive for plasma facing components (PFCs) in fusion reactor machines. An addition benefit of flowing liquid surfaces is their ability to pump gases from the divertor of a tokamak, for example. The pumping ability of flowing liquid metal surfaces is important for ash removal in the case of helium retention and low-recycling operating regimes in the case of hydrogen retention. In FLIRE, two streams of molten lithium flow down a ramp toward an exit orifice that is sealed from gas penetration by the liquid metal itself. While flowing down the ramp, an ion beam implants helium or hydrogen into the flow. After exiting this chamber, the liquid enters a separate chamber (vacuum isolated by the liquid metal) where the evolution of helium or hydrogen from the liquid is measured with a residual gas analyzer. Knowing the pumping speed implantation current and equilibrium helium/hydrogen partial pressure, the retention can be measured. FLIRE measurements of helium retention in liquid lithium show that retention varies linearly with implantation energy from 0.1% to 0.4% between 500 eV and 4keV for flow speeds of 22 cm/s over a distance of 10 cm. Based on these measurements, an effective diffusion coefficient for helium in the FLIRE lithium flow is estimated. Hydrogen retention in the flowing lithium stream is also being investigated in FLIRE. Using thermal desorption spectroscopy, the amount of hydrogen in the liquid lithium flow stream was estimated to be <0.5%, allowing an estimate of the hydrogen trapping rate.

Impact of Boundary-Layer Cutting on Free-Surface Behavior in Turbulent Liquid Sheets

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The HYLIFE-II conceptual design uses arrays of high-speed oscillating and stationary slab jets, or turbulent liquid sheets, to protect the reactor chamber first walls. A major issue in thick liquid protection is the hydrodynamic source term due to the primary turbulent breakup of the protective slab jets. During turbulent breakup, drops are continuously ejected from the surface of turbulent liquid sheets and convected into the interior of the cavity, where they can interfere with driver propagation and target injection. Experimental data for vertical turbulent sheets of water issuing downwards from nozzles of thickness (small dimension) $\delta = 1$ cm into ambient air are compared with empirical correlations at a nearly prototypical Reynolds number $Re = 1.3 \times 10^5$. A simple collection technique was used to estimate the amount of mass ejected from the jet surface. The effectiveness of boundary-layer cutting at various "depths" into the flow to reduce the source term and improve surface smoothness was evaluated. In all cases boundary-layer cutting was implemented immediately downstream of the nozzle exit. Planar laser-induced fluorescence was used to visualize the free-surface geometry of the liquid sheet in the near-field region up to 25δ downstream of the nozzle exit. Large-scale structures at the edges of the sheet, typically observed for $Re < 5.0 \times 10^4$, reappeared at $Re = 1.3 \times 10^5$ for sheets with boundary-layer cutting. The results indicate that boundary-layer cutting can be used to suppress drop formation, *i.e.* the hydrodynamic source term, for a well-conditioned jet but is not a substitute for well-designed flow conditioning.

Flow Conditioning Design in Thick Liquid Protection

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The HYLIFE-II conceptual design proposed using arrays of high-speed oscillating and stationary slab jets, or turbulent liquid sheets, to protect the reactor chamber first walls from damaging neutrons, ions and X-rays. Flow conditioning can be used to reduce turbulent fluctuations in these liquid sheets and thereby reduce surface ripple, or free-surface fluctuations, and delay jet breakup. Several flow conditioning configurations are studied experimentally for vertical turbulent sheets of water issuing downwards from nozzles of thickness (small dimension) $\delta = 1$ cm into ambient air for Reynolds numbers $Re = 5.0 \times 10^4$ and 1.3×10^5. In particular, the role of one or more fine screens in the flow conditioner was studied. As the flow conditioning element immediately upstream of the nozzle inlet, fine screens have been shown to have a major impact upon the sheet free-surface geometry. Planar laser-induced fluorescence was used to measure the free-surface geometry of the liquid sheet and its fluctuations in the near field at streamwise distances downstream of the nozzle exit $x \leq 25\delta$. Laser-Doppler velocimetry was used to quantify the impact of different conditioning configurations on the cross-stream velocity component and its fluctuations just upstream of the nozzle exit. The results indicate that minor differences in velocity and velocity fluctuations near the nozzle exit can lead to major variations in free-surface geometry, and that free-surface fluctuations are strongly affected by changes in flow conditioner design, even in the near-field region of the flow. A single screen configuration was shown to produce the smoothest jets at both Reynolds numbers, with fluctuations of 3.3% at $Re = 1.3 \times 10^5$ and $x = 25\delta$.

Visual Tsunami modeling of ballistic and assisted-pinch heavy-ion target chambers

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The Berkeley gas dynamics code TSUNAMI was recently employed to model two thick-liquid chambers: the neutralized ballistic RPD chamber [1], based on the HYLIFE-II design, and a novel assisted-pinch chamber [2]. Major revisions to the gas dynamics code TSUNAMI described in a companion paper [3] motivated a reinvestigation of previous gas transport studies. "Visual Tsunami" is therefore employed to update the modeling of the RPD chamber and the latest version of the assisted-pinch point design. Major improvements include a new real gas equation of state for the molten salt flibe, a new target output spectrum, and an improved ablation model.

Compared to previous recent simulations, new target and ablation models predict significantly more ablated mass and lower average specific energy, leading to higher impulses to thick-liquid jets. Increasing the pocket radius and/or adding more mass to the hohlraum to shift the target and hohlraum debris x-ray spectrum can mitigate this concern.

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Experimental and Numerical Investigation of Mist Cooling for the Electra Hibachi

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An experimental and numerical investigation has been conducted to examine the effectiveness of gas/liquid mist as a means of cooling the Electra hibachi structure. The aim is to quantify the effect of various operating and design parameters, *viz.* gas/liquid combination, gas velocity, liquid mass fraction, and liquid atomization nozzle design (*i.e.* spray geometry, cone angle, and droplet size distribution), on mist cooling effectiveness. The data are used to validate a mechanistic model which can be used to predict the hibachi foil's response under prototypical pulsed operating conditions.

A fully-instrumented experimental test facility has been designed and constructed. The facility includes three electrically-heated test sections, two of which are cylindrical, while the third is rectangular with prototypical Electra hibachi dimensions. Water is used as the mist liquid, with air, or helium, as the carrier gas. Three mist generating nozzles with significantly different spray characteristics are used. The carrier gas flow rate and inlet mixture conditions are varied within the expected parameter ranges dictated by circulating power constraints and the need to limit electron beam attenuation through the coolant and thin liquid film expected to form on the surface of the hibachi foils. Values of the local heat transfer coefficient along the channel surface are measured. The data indicate that mist cooling can increase the heat transfer coefficients by nearly an order of magnitude compared to forced convection using only the carrier gas. Comparison has been made between the data and predictions of a mechanistic three-dimensional computer program for transient two-phase flow in the channel coupled with heat conduction in the surrounding structure; excellent agreement has been obtained. The results indicate that gas/liquid mist can effectively cool the Electra hibachi structure within the design constraints imposed on circulating power requirements and electron beam attenuation.

Prediction of Pressure Drop in the ITER Divertor Cooling Channels

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This study is dealing with the prediction of pressure drop in the cooling channels of divertor plates in the ITER (International Thermonuclear Experimental Reactor) fusion reactor. The divertor plates in the ITER fusion reactor directly contact the plasma stream and will be exposed to extremely high heat flux. The cooling channels embedded in the divertor plates must be adequately designed to serve as an effective heat sink so that safe operation can be assured.

The divertor cooling channels were designed with header-to-header parallel channels, which consist of straight and curved tubes. The simulated typical operating conditions for the cooling system are: coolant: water; channel diameter: 0.010 ~ 0.015 m; inlet header pressure: 3.5 MPa; inlet temperature: 50°C; mass flux up to 15 Mg/m^{2}s; heat flux: up to 10 ~ 20 MW/m^2; heat transfer modes: single-phase convection and subcooled flow boiling, with a twisted tape insert to generate swirl flow in the subcooled sections.

A user-friendly thermalhydraulics simulation package was needed and developed [Yin, 1991]* to predict, based on appropriate correlations, the heat transfer and pressure drop characteristics of the system. Given the channel geometry, heat flux distribution and inlet parameters, the program will calculate the overall pressure drop from the inlet to the outlet headers and the exit pressure of a specified section. Pressure drop due to frictional, accelerational, and gravitational effects are determined

The analytical models were based on existing formulations in the literature [Collier, 1981]. The correlations selected agree well with one set of subcooled boiling data generated at moderate heat flux levels [Chen, 1985]. However, no data is available from the open literature for subcooled boiling under the high heat flux conditions of the ITER fusion reactor. The modelling should therefore be tested against such data as such data become available.

Using the basic building blocks of the pressure drop software, it is possible to construct a piping system of an arbitrary layout. Any combination of straight sections, swirl flow sections, pipe bends and inlets and outlets can be constructed from the individual subroutines discussed. However, a code is required to interpret the specified geometry, and calculate the overall pressure drop and the exit pressure of a specified segment.

In conclusion, the method and code as presented can be used for design purposes to model the pressure drop in high heat flux coolant channels under subcooled boiling conditions. The correlations selected for predicting the pressure drop agreed well with one set of published data. However, the code should be validated with data obtained at ITER operating conditions as such data become available.

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Heat transfer issues in finite element analysis of PPCS model bounding accidents

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The safety and environmental assessment of conceptual fusion power plants provides insight into physics and technology issues of fusion power generation. Estimation of temperature excursions in structures following worst-case accidents is part of these studies, performed within the European Power Plant Conceptual Study (PPCS). Bounding accidents usually assume total and unmitigated loss of coolant or coolant flow for a prolonged time, with radioactive decay as the source of heat. A new, 3D finite elements (FE) based tool, coupling the different steps of the bounding accident analysis to the same tokamak geometry, has been extensively used to conduct the neutron transport, activation and thermal analyses for all four PPCS conceptual plant models (A, B, C and D).

As power plant passive heat removal depends on conduction and radiation heat transfer, for in-vessel and vessel components, simulation of thermal transients following accidental loss of cooling requires knowledge of the material and component thermal conductivity and emissivity. The heat is ultimately removed by ways of natural convection and radiation, between the coils, vacuum vessel, and the cryostat. The presence of air, which can be stagnant or naturally circulating due to local temperature gradients, may be assumed to fill the space between the vessel/coils and cryostat, but conservative analyses have also been performed with vacuum conditions.

This paper presents results obtained relating to the effects on thermal performance of PPCS plant models during bounding accidents, for the following conditions:

(a) The effect of radiation heat exchange between the inner surfaces of the tokamak, i.e. the first wall and divertor, is examined. As the divertor sustains high temperatures during an accident, radiation exchange with the surrounding structures is important.

(b) The effect of stagnant or naturally circulating air, on the heat removal and resulting temperatures is assessed. In general it is demonstrated that the temperature differences of various parts of the structure set in motion a circulation pattern which facilitates heat flow to the surrounding cryostat and eventually the atmosphere.

(c) The ability of the blanket to remove heat by conduction is partly dependent on the thermal conductivity of the multiplier and tritium generating materials. A general feature of these materials is the degradation of thermal conductivity with exposure to a neutron environment. The effect of different values on the heat transfer capability of the blanket is examined and discussed.

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Experimental Research on Heat Transfer Enhancement for High Prandtl-Number Fluid

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A molten salt Flibe is one candidate material for fusion blanket coolant/breeder since Flibe can play a role in breeding tritium and cooling. The Flibe has many advantages for blanket material. However, the Flibe has also several disadvantages, especially high Prandtl number due to its large viscosity. Generally, a thermal boundary layer of a high Prandtl-number fluid is thin in comparison with a velocity boundary layer. This thin thermal boundary layer causes a low heat-transfer coefficient. Therefore, using the Flibe as a coolant for high surface heat flux conditions, we have to develop a new method of heat-transfer enhancement in a Flibe channel flow to stir the thin thermal boundary layer by some active promoters since turbulent heat-transfer enhancement wouldn't work for high Prandtl-number fluid.

The experimental research on heat-transfer enhancement has been performed with a large molten salt circulating experimental loop that is called "TNT loop" (Tohoku-NIFS Thermofluid loop). Although the TNT loop was designed and constructed in order to employ the Flibe as a working fluid, Heat Transfer Salt (HTS) has been used as a simulant of the Flibe since the melting point of Flibe is so high and its vapor contains beryllium. The compositions of HTS are $NaNO_3$, $NaNO_2$ and KNO_3. With the HTS, we can achieve the same Prandtl number as that of Flibe under lower temperature.

We have employed packed-bed tube as the enhancer for molten salt since packed bed can stir the thin thermal boundary layer and fluid flow and so on. Stainless-steel bed and copper bed are chosen as the enhancer to evaluate the effect of material of bed. Especially, in case of the stainless-steel bed, several diameters of metallic spheres are changed as parameter; 1/2 and 1/4 of diameter of circular tube. In the copper bed, the diameter of spheres is fixed to be 1/4 of the tube diameter.

Through the experiments, it is clarified that the enhancement of packed-bed tube is superior to that of turbulent heat transfer from the viewpoint of the same flow rate. Also, the 1/4-diameter bed is superior to the 1/2-diameter one at the same flow rate. Furthermore, at low flow rate, small differences of heat transfer performance can be seen between the stainless-steel and copper beds. At high flow rate, however, the heat-transfer coefficient ratio strongly depends on the flow rate in the case of the 1/4-diameter copper bed which can't be observed in the case of the 1/4-diameter stainless-steel bed. As a result, it is considered that the thermal energy is expanded from a heated wall deeply through the packed beds at low flow rate, so that the difference of thermal conductivity of metallic sphere has little effect for the total heat-transfer enhancement. On the contrary, it is also considered that the convective heat transfer in the vicinity of a heated wall is strong at high flow rate, so that the difference of the conductivity is effective.

The evaluation from the viewpoint of the pressure drop shows that the turbulent heat transfer is superior to packed-bed tube. However, the heat transfer of packed-bed tube is not proportional to the pressure drop at low flow rate. This means that the most optimum thermofluid flow exists where the enhancement of packed-bed tube is better than that by the turbulent flow.

Initial Study of Supercritical Fluid Blowdown

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Improved efficiency of fusion power plants can be achieved with higher operating temperatures and pressures. By taking advantage of the large change in the specific heat at the pseudo critical point of fluids, fusion power plants could gain efficiency improvements as high as 30% in the secondary side. There are however several issues that need to be resolved including possible accident situations where a tube or pipe rupture ensue fluid from the break. This would result in a rapid depressurization of the system governed by the critical flow rate from the tube rupture. The paper first presents a brief review of the previous work done on critical flow of supercritical water (SCW) by United Kingdom Atomic Energy Authority (UKAEA) and concerning supercritical CO_2 depressurization of vessels by the University of Hamburg. Then a model based on a steady state Homogeneous Equilibrium Model (HEM) and conditions with and without friction is also presented. Calculations indicating three different possible regimes in a blow down scenario are calculated with this model. The single-phase flow in the supercritical region and the transition either in to sub-cooled water, a two-phase fluid or a superheated gas near the critical point results in an interesting flow with a wide range of behavior. Indeed depending on the initial conditions and the geometry either vaporization or condensation can occur either in the pipe or at the exit. A comparison between the results and the experimental data from UKAEA and University of Hamburg-Harburg is discussed. Moreover those results are attempted to be extended to other fluids like CO_2 , R22 or R134a by comparing its thermodynamic and dynamic evolution to dimensionless SCW results. Finally experiments with the depressurization of a supercritical water system are presented and designs of an experimental apparatus to observe this phenomenon during a pseudo steady state or a transient blowdown with several different fluids is discussed.

Evaluation of flow structure in packed-bed tube by visualization experiment

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Liquid blanket system in a nuclear fusion reactor has a role of generating tritium and transporting the fusion energy. Molten salt Flibe, which is a mixture of LiF and BeF_2, is one of the candidates for the liquid blanket material. The Flibe blanket system, however, needs heat transfer enhancement device under high heat flux because the Flibe is a high Prandtl number fluid. The packed-bed tube, which is filled with a number of spheres, is one of the heat transfer enhancement techniques. The heat transfer characteristics of packed bed tube have been evaluated by using Heat Transfer Salt (HTS) as the Flibe simulant in Tohoku-NIFS Thermofluid (TNT) loop at Tohoku University. [1]

The heat transfer enhancement performance of the packed-bed tube is determined by the filling structure of spheres, the material of spheres, the flow structure and properties of the fluid. Since the mechanism is not clarified how the flow structure affects the heat transfer enhancement and pressure drop characteristics, it is necessary to evaluate the flow structure in the packed-bed tube. The purpose of this study is, therefore, to investigate the flow structure in the packed-bed tubes through visualizing the flow field by a PIV system.

The test section of experimental apparatus is made of an acrylic tube whose inner diameter is 56mm and is filled with acrylic spheres with the diameter of 27.7mm and 18.5mm. In order to measure the pressure drop in the test section, the Bourdon tube pressure gage is installed in the upstream and the downstream sides of the test section respectively.

Generally, the visualization inside the packed-bed tube is difficult due to the existence of spheres. It is, therefore, necessary to match the refractive index of working fluid with that of spheres to visualize the whole flow field in the packed-bed tube. It is confirmed that the refractive index of sodium iodide solution (NaI) with the concentration of 61-62% at the temperature of 298K corresponds to that of acrylic spheres. Through the experiment using this NaI fluid, the flow structure in the packed-bed tube can be visualized by the PIV system. It is clarified that the flow structure near the tube-wall is strongly influenced by the flow in the center area of tube. The pressure drop shows almost the same tendency given by the experimental formula of Ergun. [2]

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Heat Transfer Enhancement Technique with Copper Fiber Porous Media

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In a fusion reactor, almost 30% of fusion energy is deposited on plasma facing components. In the diverter region, it is, however, difficult to utilize this energy with conventional cooling techniques based on high velocity flow with highly subcooled cooling. From this viewpoint, the authors have been developing a cooling technique with metal porous media, and have succeeded in removing inlet heat flux of over $50MW/m^2$. In this study, in order to attain both the higher cooling performance and the acquisition of high-energy density, high heat removal experiments are performed by using cylindrical homogeneous and functionally graded metal porous media to estimate their fundamental heat transfer performances. The sintered porous media made of copper fibers are used with water coolant flowing through the porous media.

From the experiments with the homogeneous porous media whose porosity and fiber-diameter are spatially uniform, it is verified that the cooling performance of porous media becomes higher as the pore size is smaller. It is also clarified that the heat transfer mechanisms including phase change can be classified into three patterns depending on the flow velocity of coolant. In these regions, there exists an optimum flow rate in which it becomes possible to achieve both the high cooling performance and the high outlet temperature simultaneously.

On the other hand, as for the porous media with the finer pore, the pressure loss becomes higher, which prevents the increase of flow velocity of cooling water. For this reason, the heat transfer enhancement by forced-convection is quite low. To overcome this disadvantage, the heat transfer experiments are carried out by using functionally graded porous media whose pore size changes along the longitudinal direction.

The functionally graded porous media can reduce the pressure loss. In case of the porous media with the fine pore, the heat transfer coefficient is higher than that obtained in homogeneous case under the low inlet-pressure condition. As for the outlet temperature, the difference is not seen between the functionally graded and the homogeneous porous media. This result indicates that the evaporated ratio of water coolant increases, while the steam temperature is not raised.

Thermal Analysis of the Z-Pinch Power Plant Concept*

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The Z-Pinch Power Plant (ZP-3) is the first concept to use the results at Sandia National Laboratories' Z accelerator in a power plant application. Assuming high yield fusion pulses of 1 to 20 gigajoules per shot at a rate of 0.1 Hz, a unique shock and energy absorbing system is being considered to contain and utilize the energy released. This integrated blanket shields the permanent structures from the high-energy neutron flux and the strong shock wave, breeds tritium, and is the working fluid that absorbs the released fusion energy. The absorbed fusion energy is then utilized to drive a power cycle to produce electrical energy. The total electrical power output of the Z-Pinch Power Plant considered is in the range of the power generated by a conventional, 1 GWe power plant..

In this paper, we report on the results of detailed thermal analyses of the ZP-3 power plant. In particular, we evaluate the transient heat up and steady-state temperature fluctuations of the fusion reactor as well as the thermal efficiency of various power generation schemes. For this preliminary study, the power generation schemes considered are variations of the standard Rankine and Brayton cycles. A focus of the study is to determine the impact of various working fluids on the time to reach steady state temperature, the peak operating temperature, the expected range of temperature fluctuations, the required working fluid mass flow rate, as well as the thermal efficiency and temperature differential of the electrical power generation scheme. All working fluids considered contain enough lithium to breed the required tritium fuel and include pure lithium and molten lithium fluoride salts with various amounts of beryllium and sodium.

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Modeling Development for Free Surface Flow with Phase Change

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Development of predictive capability for free surface flow with phase change is essential to evaluate liquid wall protection schemes for various fusion chambers. Wtih inertial fusion energy (IFE) concepts, such as HYLIFE-II, rapid condensation into cold liquid surfaces is required when using liquid curtains for protecting reactor walls from blasts and intense neutron radiation. With magnetic fusion energy (MFE) concepts, droplets are injected into the free surface of the liquid to minimize evaporation by minimizing the surface temperature. This paper presents a numerical methodology for free surface flow with heat and mass transfer to help resolve feasibility issues encountered in the aforementioned fusion engineering fields, especially spray droplet condensation efficiency in IFE and droplet heat and mass transfer enhancement on free surface liquid divertors in MFE.

The numerical methodology is being conducted within the frame work of the incompressible flow with heat and mass transfer model. We present a new second order projection method, in conjunction with Approximate-Factorization techniques (AF method), for incompressible Navier-Stokes equations. A smoothing function is introduced for the piecewise constant density, viscosity and temperature. The Crank-Nicholson method was used for the diffusion term to eliminate the numerical viscous stability restriction and 3rd order ENO scheme used for the convective term to guarantee the accuracy of the method. A four-level V cycle multigrid algorithm for pressure Poisson equation is used in order to decrease computation time. To capture the free surface of the flow and the deformation of the droplets accurately, we use the level set method by S. Osher and J.A. Sethian. The level set approach has two inherent strengths. One very useful feature is that the representation of the interface as the level set of some function ϕ leads to convenient formulas for the interface normal direction and curvature. Another advantage of this approach is that no special procedures are required in order to model topological changes of the front. This numerical investigation identifies the physics characterizing transient heat and mass transfer of the droplet and the free surface flow. The results show that with the deformation of the droplet and topological changes of the free surface, the heat and mass transfer can be enhanced significantly.

Experimental & Numerical Study of Ceramic Breeder Pebble Bed Thermal Deformation Behavior

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The objective of this paper is to develop predictive capability for pebble bed deformation. The phenomena and parameters involved in ceramic breeder pebble bed thermal deformation are complex. First, unlike solid material, stress distribution in a pebble bed system is highly concentrated at the local zone of contact area. Second, deformation behavior is time-dependent.

According to previous studies[1,2], the initial strain rate can be significant, but as the contact area grows, it can become much relaxed and simultaneously stress magnitude is reduced. Therefore, deformation mechanisms at contact points in the pebble bed system are different and evolve with time. Deformation in the pebble bed may be detrimental and, therefore, understanding its behavior is important to ensure adequate blanket performance.

The experimental facility is built and operated under reactor-like temperature operating conditions (500-800^{o}C). The test pebble bed structure represents a simplified model of a typical solid breeder blanket system, which provides data of fundamental thermal deformation behavior. The recorded data includes the deformation magnitudes of interaction between the structural wall and pebble bed at different times and temperatures. Numerical simulation based on finite element method (FEM) is employed to study details of the deformation. Constitutive equations of the creep model derived experimentally by Bühler[3] have been incorporated into the numerical model. The preliminary results show that the simulation is able to capture thermal deformation characteristics. However, the absolute values still need to be resolved using correct material properties.

[1] J. Reimann and G. Wörner, “Thermal Creep of Ceramic Breeder Pebble Beds”, *“Proceedings CBBI-9”*, Toki (2000)
[2] J. An, A.Y. Ying and M. Abdou, Progress on Experimental & Numerical Study on Ceramic Breeder Pebble Bed Thermal Creep Behavior, CBBI-11, Japan, 2003
[3] L. Bühler, Continuum models for pebble beds in fusion blankets, FZKA 6561 (2002)

Effects of Pulsed Operation Conditions on Effective Thermo-physical Properties of Ceramic Breeder Pebble Beds

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Using ceramic breeder pebble beds inside the fusion blanket is a leading and promising concept in many fusion studies and blanket designs. The heat transfer parameters of the pebble beds, namely effective thermal conductivity and interface thermal conductance, control the bed's thermal performance. For a typical design of a ceramic breeding blanket, the temperature window of ceramic breeder was selected to range from 400°C to 900°C. Studying the thermo-physical properties of ceramic pebble beds under this high temperature range is a challenge to any experimental work. An advanced high temperature experiment was designed and built at UCLA to study the thermal performance of ceramic breeder pebble beds under pulsed operational conditions and the aforementioned high temperature window.

This work presents an experimental study of effective thermal conductivity, k_{eff}, of a ceramic breeder (Lithium Titanate) pebble bed and the interface heat conductance, h, between the ceramic pebbles and the surrounding structural wall. In addition these parameters (k_{eff} & h) were investigated under pulsed thermal loads in order to understand and quantify the effects of pulsed operation conditions on these parameters. The behavior of pebble beds may not show the same thermal behavior after loading with a specific number of thermal cycles. The objective is to determine how the cyclic thermal effects will degrade and/or change the thermal performance of the pebble bed. Also, finite element analysis (using ANSYS in 3D mode) was carried out to compare the thermal performance of the pebble bed with the experimental results.

Magnetohydrodynamic Turbulent Channel Flow with Transverse Square Cylinders

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Turbulent flows of electrically conducting fluids are of most importance in the various engineering aspects of the future fusion power plant. Some of these flows are expected to interact directly with immersed solid bodies. Although a significant number of direct numerical simulations (DNS) have been reported for magneto-hydrodynamic (MHD) turbulent flow in channels, only few investigations taking into account such bodies have been considered so far.

Direct Numerical Simulations of MHD turbulence in channel flow with transverse square cylinders placed at selected positions normal to the channel wall are presented. The magnetic field is assumed to be parallel to the stream-wise direction of the flow. This external magnetic field configuration was selected because it minimizes the Hartmann effect on the non-conducting walls of the channel, although a strong Hartmann effect is encountered at the cylinder walls. The induced magnetic field is assumed to be negligible and, thus, the low magnetic Reynolds number approximation has been used, instead of the full magnetic induction equations. The numerical method used to solve the low-Rm equations is based on a semi-implicit fractional step method; the diffusion term is advanced in time with the Crank-Nicolson method, while the non-linear and Lorentz force terms are advanced with a third-order Runge-Kutta (RK3) method. An efficient immersed-boundary method is adopted for the square cylinders.

The parameters studied in the present work are the magnetic interaction parameter (Stuart number), which is the ratio of the electromagnetic force to the inertial force, and the size and position of the transverse square cylinders. The Reynolds number is fixed at the relatively low value of 3000. The results show that the usual MHD turbulent channel flow is characterized by more organized and elongated structures than that in the channel flow without a magnetic field. The streaky structures are substantially weakened and the streak spacing appears to be larger. The influence of the simultaneous presence of the transverse square cylinders with the magnetic field is investigated. Results show a strong connection of the Stuart number with the Hartmann effect due to the presence of the square cylinders.

MHD Effects on Heat Transfer in a Molten Salt Blanket

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We consider heat transfer in closed channel flows of molten salts under specific reactor conditions as applied to a blanket concept. Either Flibe or Flinabe is used as a coolant and tritium breeder. As an example of such a concept we can refer to a re-circulating blanket [1]. Several dual-coolant blanket options [2] are also currently accessed as possible blanket candidates in the US DEMO plant. Since the electrical conductivity of molten salts is relatively high (~10^2 1/Ohm-m), such flows experience non-negligible MHD interaction, which may result in significant changes in both flow and heat transfer conditions.

In the self-cooled blanket, the molten salt flows first through the cooling FW channels. The magnetic field causes turbulence reduction and heat transfer degradation. This effect depends on the direction of the applied magnetic field with respect to the main flow. The calculations for such flows have been conducted using the "K-epsilon" model of MHD turbulence [3] for two possible channel orientations, poloidal and toroidal. The analyzed conditions address Flibe flows in a 8 m channel, the velocity 5-15 m/s, the magnetic field 0-10 T, and the surface heat flux 1 MW/m^2. Depending on the flow conditions the heat transfer degradation due to the magnetic field effect (in terms of the increase of the film ΔT) can be as high as 30%.

In the blanket itself, the molten salt flows slowly (~0.1 m/s) through a poloidal channel with a typical cross-sectional dimension of 0.5 m absorbing the bulk heat generated by neutrons. The flow regime is laminar due to turbulence suppression by a magnetic field. One of the design requirements is to reduce the amount of heat escaping from the molten salt into the cooling helium channels in the surrounding structure. The calculations were conducted using a 3-D heat transfer model. The velocity profile was calculated with a 2-D MHD code and then used as the input data in the heat transfer analysis. The calculated data show that under the DEMO plant conditions [2] no more than 15% of all heat generated in the Flibe escapes into the helium. Another important conclusion is that the temperature at the interface with the ferritic structure does not exceed the maximum allowable temperature of 550°C.

1. C.Wong *et al.*, *Molten salt self-cooled solid first wall and blanket design based on advanced ferritic steel*, to appear in FED.
2. C.Wong *et al.*, *Assessement of liquid breeder first wall and blanket option for the DEMO design*, TOFE abstract.
3. S.Smolentsev *et al.*, *Application of the K-epsilon model to open channel flows in a magnetic field*, International Journal of Engineering Science, 40, 693-711 (2002).

Study of Liquid Metal Film Flow Characteristics under Fusion Relevant Magnetic Fields

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The use of fast flowing liquid metal thin films as the divertor surface in a magnetic fusion device is a very attractive option for effective particle pumping and surface heat removal. The strong magnetic field environment tends to create flow disrupting magneto hydro dynamic (MHD) forces, which pose a major challenge in establishing a smooth, predictable flow of liquid metal over the divertor surface. The present study builds up on the ongoing research effort at UCLA, directed towards providing qualitative and quantitative data on liquid metal film flow behavior under NSTX [1] relevant magnetic field conditions and identifying design constraints for implementation of a liquid surface divertor module.

The study is being conducted at the magnetic torus (M-TOR) facility at UCLA, with capabilities to reproduce a scaled toroidal NSTX magnetic field environment. A Gallium alloy (Ga-In-Sn) is being used as the liquid metal. Preliminary experiments have been conducted using a 34 cm long and 5 cm wide stainless steel channel. The flow behavior has been qualitatively and quantitatively studied under different conditions of applied magnetic field and average fluid velocity. These experiments have lead to several interesting insights, most significant of them being a three to four times increase in the local film thickness as the flow progresses downstream, with a decrease in the average flow velocity by about 35%. It has also been observed that different magnetic field components have very different effect on the flow behavior. In order to build up on these preliminary findings, two new test sections have been designed. The first is a modification of the conducting channel being currently used, with added features of increased flow length of 42 cm and a stream-wise expansion of the channel span (from 5 cm at inboard to 7 cm at outboard). The second test section features a ‘wide channel’ flow of liquid metal (using a 20 cm wide conducting channel as opposed to the 5 cm wide current channel) under an applied surface normal magnetic field component, with a positive spatial gradient in the flow direction. This will lead to a better understanding of the flow behavior under an applied surface normal magnetic field component, while mitigating the hydrodynamic ‘wall’ effects. In order to supplement the experiments a conscious numerical modeling effort has been started. A 3D incompressible MHD free surface fluid modeling code called ‘HIMAG’ originally developed by HyPerComp Inc. is being modified for the problem at hand and will be used to get further insights into the physical phenomenon. Results from the new experiments and numerical modeling will be duly reported.

1: National Spherical Torus Experiment, an experimental fusion device being operated by the Princeton Plasma Physics Lab

Oral Session II-4

US Contributions to ITER - Special Session

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The ITER CS Magnet System

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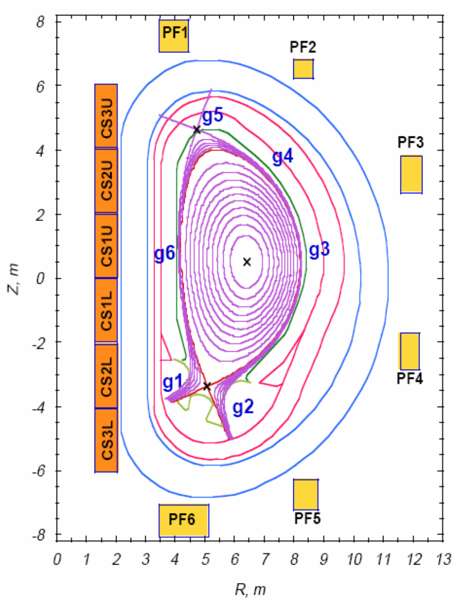
ITER is planned as a high Q, extended burn tokamak, with two main experimental goals: (1) a long inductive burn with Q=10 and (2) a steady-state non-inductive burn at Q=5. In addition, the ITER mission includes the integration of fusion technologies, power reactor component testing, and exploration of tritium breeding concepts. As an experimental device with such a broad mission, operational flexibility and extensive diagnostics will be required. To span these burning plasma requirements, active profile control will be required- especially in the gaps g1-g6 in the figure below. Plasma confinement and shaping are largely controlled by adjustment of the toroidal and poloidal field coils and the evolution of the plasma current. The planned long pulse length, of order 300-500s, dictates the use of superconductors in all of the coil systems. To approach reactor-like conditions, high fusion power (with an associated β^2B^4 scaling) is desired. The D-shaped TF magnet coils, shown in elevation, are pushed to the highest possible toroidal field, limited by the maximum allowed conductor field. The TF coils use high performance Nb_3Sn conductors. The outer PF coils, intended to provide plasma shaping and active plasma control, are large and driven more by cost than performance, and will use NbTi conductors. In the inductive heating mode at high Q, the initial heating is obtained by a large flux swing in the central solenoid, and a rapid and complete field inversion from +13.5 T to -12T is required. The CS coils will also employ high performance Nb_3Sn conductors. All coils use cable-in-conduit-conductor (CICC) windings.



On the basis of this key plasma performance role, prior US experience and other factors, the US has proposed to supply all or part of the CS magnet system to ITER, with the US having the technical lead. The post 1998 ITER rescaling has altered the CS design basis from the tested EDA CS model coil. Recent developments have further changed the design. The CS consists of 6 identical free standing modules, stacked and compressed into a support structure mounted on the upper TF inter-coil cases. The CS no longer bucks the TF in-board legs. Each CS module is assembled from 7 sub-module windings, each of which uses 800m CICC conductor lengths. Module fabrication is wind-and-react followed by impregnation. A number of activities are in progress in the US; centered on baseline design confirmation and risk mitigation, as well as pre-production and industrial scale up planning. The present CS design and performance, key engineering issues, current activities, and the US planning for the supply of the CS are presented.

Proposed US Participation in Fabrication of the First Wall and Shield for ITER*

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As a result of international negotiations on the division of responsibility for providing components to the International Thermonuclear Experimental Reactor (ITER), the US has tentatively agreed to provide one toroidal ring of the first wall and shield (FWS) modules. The US portion is the ring just above the outer divertor (module 18 in ITER notation). There are 36 modules in the ring and 4 spare modules are to be provided. Each module is approximately 1.2 by 1.3 m by 0.4 m (thick) and weighs about 3.6 Tonnes. The plasma facing surface is about 10% of the first wall area. Each first wall module consists of 10 mm of Be joined to 11 mm of CuCrZr alloy bonded to 59 mm 316LN stainless steel with two rows of 12 mm diameter water cooling channels. The first wall is mechanically attached to the shield module. Even though there are not any detailed manufacturing drawings of module 18, we have estimated the cost of the modules and the research and development (R&D) needed to successfully fabricate the modules to meet ITER requirements. We will discuss the cost estimation basis. We will work with the ITER Team at Garching, Germany to develop the final manufacturing drawings and specifications.

The R&D identified falls into four general categories. Based on results from the Engineering Design Activity (EDA), we are considering the use of plasma spray to place the Be on the copper alloy but we need to optimize the process for the FWS geometry and requirements. The copper alloy is a precipitation hardened alloy and techniques for joining the copper to stainless steel must be done at a temperature less than about 500C to prevent reduction of the copper alloy strength or else a complex solutionizing, quenching, and precipitation process must be performed to regain the copper properties. Another complication is to avoid formation of brittle Be/Cu intermetallics that are formed at high temperature. We plan to conduct R&D on this issue. Our success with using casting as a cost effective means for fabricating large 316LN structures during the EDA leads us to consider such a fabrication technique for the shield module. Casting offers the possibility of casting the module with the internal cooling channels in place rather than forming the channels by machining. We will investigate the possibility of using casting to fabricate the shield module. Finally, techniques must be found for assuring the thermal bonds in the structure are adequate to remove the plasma heat load on the first wall at a sufficiently early stage in the manufacturing to assure rejection of non-conforming parts before completion of all steps to avoid excessive costs due to non-conforming parts. We will conduct studies to determine the best methods for quality control and quality assurance during the R&D process to ensure that we have suitable processes for the final manufacturing.

* Sandia is a multi-program laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

ITER Ion Cyclotron Heating and Fueling Systems

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The ITER ion cyclotron system offers significant technology challenges. The antenna must operate in a nuclear environment and withstand heat loads and disruption forces beyond present-day designs. It must operate for long pulse lengths and be highly reliable, delivering power to a plasma load with properties that will change throughout the discharge. A development and testing program will be required to validate the proposed ITER antenna design, and to modify it if needed.

The ITER ion cyclotron system consists of one eight-strap antenna mounted in a horizontal midplane port; eight rf sources covering the 35-65 MHz frequency range that can deliver a total of 20 MW of IC power to the plasma, and associated high-voltage DC power supplies; and a set of transmission lines connecting the antenna to the sources, with matching and decoupling components.

There are several R&D tasks that must be completed successfully before the final procurements and fabrication can be started. A low-power electrical mockup of the antenna must be fabricated and tested in order to validate the functioning of the antenna and to optimize the design. A prototype of the tuning mechanism must be designed and tested. A high-power prototype antenna must be built and tested under vacuum at voltages similar to those expected for ITER operation. This prototype will be one current strap (i.e., 1/8 of the full antenna) with the tuning mechanisms, vacuum transmission line and vacuum window. A test of the proposed dual-output transmitter tube configuration with a variable load that simulates load changes during plasma operation will be required.

The ITER fueling system consists of a gas injection system and multiple pellet injectors for edge fueling and deep core fueling. Pellet injection will be the primary ITER fuel delivery system. The fueling requirements will require significant improvements in pellet injector performance. The pellet injectors must safely operate with tritium and be able to reliably supply hydrogenic species throughputs well beyond present-day designs. It must operate for long pulse lengths (~3000 s) and be highly reliable, delivering nearly 400 torr-L/s of tritium rich pellets. The proposed design is based on a centrifuge accelerator fed by a continuous screw extruder. Inner wall pellet injection with the use of curved guide tubes will be utilized for deep fueling.

A development and testing program will be required to validate the proposed ITER pellet injector design, and to modify it if needed. A high throughput extruder prototype would be tested in the laboratory. A centrifuge prototype (capable of producing 3mm pellets at 50 Hz at a speed of ≥300 m/s) would be tested in the laboratory and field tested on an existing fusion device.

ITER ECH System and US ECH Program for ITER

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The Electron Cyclotron Heating and Current Drive (ECH / ECCD) system for ITER has a goal of:

- EC Current Drive (ECCD), on-axis and off-axis.
- EC Heating (ECH), including start-up.
- Neoclassical Tearing Mode (NTM) stabilization.

To accomplish this goal, the ECH system consists of the following five subsystems:

- 24, 1 MW 170 GHz Gyrotron Systems and 3, 1 MW 120 GHz Gyrotron Systems.
- DC Power Supplies and Controls for the Gyrotrons.
- Transmission Line System.
- 1 Equatorial Plane Launcher.
- 3 Upper Port Launchers.

The US ECH community has begun work on possible US contributions to the ITER ECH system. The US is interested in contributing in all areas of ECH technology. The US has completed a study of the 1 MW, 170 GHz gyrotrons and has produced a successful short pulse (3 microseconds) prototype. Work is being initiated on the design of a 1 MW, 120 GHz gyrotron, which would operate with very high reliability. Research is also underway on a remote, steerable launcher, which could contribute to the goal of NTM stabilization. An update will be given of the US program for research and development of components for the ITER ECH system and US plans for contributing hardware to the ITER project.

Future US ITER Safety Studies

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The US Fusion Safety Program played a prominent role in the development of safety related data and documents during the Engineering Design Activity (EDA) phase of the International Thermonuclear Experimental Reactor (ITER) project. This role included defining possible accident scenarios for ITER based on existing failure rate data given the adopted radioactive confinement strategy for ITER, and the analysis of these accident scenarios using aerosol data, tritium permeation data, and safety related computer codes. The culmination of this safety activity was the publication of the ITER Non-Site Specific Safety Report (NSSR). ITER safety work has continued during the ITER-FEAT design activity, in the absence of US participation but with the use of US developed safety analysis tools, to produce the ITER-FEAT Generic Site Safety Report. Now that the US has reentered the ITER project, the US Safety Program has been tasked to address safety areas left unresolved by the absence of the US during the past five years. There are four general areas that will be addressed, which are:

1) validation of US safety analysis tools (calculations with quantified uncertainties with the level of detail to depend on regulatory requirements of the actual site) that underpin the ITER safety analysis,
2) validation of magnet safety codes against medium scale magnet and busbar arcing experiments to demonstrate that ITER can tolerate large internal and external arcs in the magnet systems without degrading the integrity of radioactive confinement barriers to the point where resulting radioactive releases produce site boundary doses that exceed allowable limits,
3) refinement of dust inventory estimates in ITER, development of dust mobilization data under off-normal conditions and development of a dust removal strategy that will demonstrate compliance with dust safety limits in ITER and not hamper operational flexibility of the machine, and
4) further refinement of tritium inventory estimates in ITER mixed material plasma facing components (PFCs) and demonstration that removal strategies are effective at the ITER scale to comply with safety limits.

In this paper we describe these task areas and the time frame for completing these tasks.

US Tritium Plant Activities for ITER

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The US has contributed to ITER Tritium Plant activities since the beginning of the project. Initial contributions were made to the Conceptual Design Activities in the late 1980's. Both R&D and design contributions were made to the Engineering Design Activity in the 1990's. As ITER now moves to construction, the US is slated to build and deliver the Tokamak Exhaust Processing (TEP) system. The main purpose of this system is to recover hydrogen isotopes from molecules such as water and methane, and deliver purified hydrogen isotopes to the isotope separation system. The TEP construction activity will begin with finalizing the detailed design. Then industry will fabricate the system. This will be followed with factory testing, transportation to the ITER site, installation and final acceptance testing. This system is highly integrated with other Tritium Plant subsystems, so close interactions are expected with other procurement package owners and the ITER central team

Oral Session II-5

Breeding Blanket Development

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Development of Solid Breeder Blanket at JAERI

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Japan Atomic Energy Research Institute (JAERI) has been assigned as a leading institute for developing the solid breeder blanket in the long-term research program of fusion blankets in Japan, which was approved by the Fusion Council of Japan in 1999. In accordance with the long term research program, element technology development of solid blanket has been performed at JAERI and showed significant progress. Based on the achievement of the element technology development, the development phase is now stepping further to the engineering development phase. This paper presents the major achievements of the element technology development of solid breeder blanket in JAERI.

According to the reactor designs of a fusion power demonstration plant in JAERI, the blanket is required to withstand the neutron wall loading of up to 5 MW/m^2, and the surface heat flux of up to 1 MW/m^2. Target availability of operation is 75%. In the fusion power demonstration plant, it is required to show the feasibility of tritium breeding ratio to be more than 1.05 and thermal efficiency of power plant system to be comparable or higher than 35%. As a near term blanket option, reduced activation ferritic steel, F82H, and high pressure water (15 or 25 MPa) were selected as the candidate structural material and coolant. To realize the selected solid breeder blanket design, the element technology development consists of the development of module fabrication technology, the development of irradiation technology for in-pile mockup irradiation test, the development of fabrication technology for breeder and multiplier pebbles, irradiation tests of breeder and multiplier pebble beds, neutronics studies on a blanket module mockup and the development of tritium recovery system.

As for the blanket box fabrication technology, treatment conditions of a Hot Isostatic Pressing (HIP) joining were studied. It was made clear that the coarse grained microstructure after conventional HIP process was refined by the post HIP normalizing process at the temperature below 1313 K. This result implies the thermal hysteresis effects could be canceled by an appropriate heat treatment. As for the evaluation of thermal and mechanical characteristics of the pebble beds, new measurement apparatus was established and the data on the effect of compressive strain on the pebble bed thermal conductivity was measured in good accuracy. This apparatus is expected to clarify effective thermal conductivity of pebble beds under creep conditions and cyclic stress conditions.

Based on the achievements of above stated element technologies, the R&D program is now stepping to the engineering testing phase, in which scalable mockup tests will be performed for obtaining engineering data unique to the specific structure of the components, with the objective to define the fabrication specification of test blanket modules for ITER.

EU Blanket Design Activities and Neutronics Support Efforts

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An overview is provided of the design activities conducted in the European Union for the development of breeder blankets for future fusion power reactors. The focus is on the related neutronics support efforts.

The European Fusion Technology Programme considers two development lines of a breeding blanket: one employing a Lithium ceramics as breeder and the other the Lithium–lead liquid metal alloy. To minimise the development costs, the blanket design and the related R&D efforts have been recently based on the use of the same coolant (high pressure Helium gas) and the same modular blanket structure, a box with a stiffening grid that can withstand the high pressure of the coolant gas in the case of an internal leak. With this design concept, the open cells of the stiffening grid accommodate small breeder units. The Helium-Cooled Pebble Bed (HCPB) blanket makes use of Li_4SiO_4 as breeder material and beryllium as neutron multiplier in the form of pebble beds. The breeder unit of the HCPB type consists of a back plate with two breeder canisters attached, each providing space for two shallow ceramic breeder pebble beds. The space between the canisters and the stiffening plates is filled with Beryllium pebbles. The Helium-Cooled Lithium-Lead (HCLL) employs the Pb-Li eutectic alloy as breeder and neutron multiplier. Breeder units of the HCLL type consist of a few cooling plates and the backplate with the Helium inlet/outlet manifolds. The Lithium-lead fills the entire space between the cooling and the stiffening plates.

The neutronic support efforts include design analyses for the layout and optimization of the modular HCPB/HCLL blankets based on detailed three-dimensional Monte Carlo calculations with the MCNP code. The related power reactor models were devised on the basis of parameters specified in the European Power Plant Conceptual Study (PPCS). The major neutronics features of the modular HCPB/HCLL blanket concepts will be presented in the paper and the main neutronics results will be discussed.

The neutronics support efforts in the European Union also include the development works conducted in the frame of the European Fusion and Activation File (EFF/EAF) projects. The main objective of these activities is to provide well qualified nuclear data and computational tools as required for reliable neutronics design calculations. This includes the development of nuclear cross-section data for neutron transport and activation calculations and their validation through benchmark experiments. The paper presents a brief overview of these activities including the neutronics benchmark experiment on a HCPB breeder blanket mock-up which is currently under preparation at the Frascati Neutron Generator.

Assessment of Liquid Breeder First Wall and Blanket Options for the DEMO Design*

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As candidate blankets for an US DEMO power plant, we assessed first wall and blanket concepts based on the use of reduced activation ferritic steel (RAFS) as structural material and liquid breeder as the coolant and tritium breeder. The liquid breeder choice includes the eutectic lead-lithium alloy Pb-17Li and low melting point molten salts such as $LiBeF_3$ with a melting point of 380°C and FLiNaBe with a melting point of ~305+15°C. Molten salt blankets require additional neutron multiplier like Be to provide adequate tritium breeding. In order to meet the temperature limitation of RAFS with T_{max} < 550°C, while getting high coolant outlet temperature for high thermal efficiency, we selected the dual coolant configuration for our designs. Helium is used to remove the first wall surface heat flux and to cool the entire steel structure. The liquid breeder is circulated to external heat exchangers to extract the heat from the breeding zone (a "self-cooled" breeding zone). We take advantage of the molten salt low electrical and thermal conductivity to minimize impacts from the MHD effect and the heat losses from the breeder to the helium cooled steel structure. For the Pb-17Li breeder we employ flow channel inserts with a low electrical and thermal conductivity to perform similar functions. We based our first wall and blanket assessment on a DEMO design, which has a maximum neutron wall loading of 3 MW/m^2 and a maximum surface heat flux of 0.42 MW/m^2 at the outboard midplane of the tokamak reactor. This paper reports on the status and results of our assessment, including the logic behind materials and design configuration choices. Preliminary analyses including neutronics, thermal-hydraulics, thermal-mechanics, safety, tritium-control and power conversion system are presented. R&D items are also identified, which form the technical basis for the formulation of the US ITER test module program and corresponding test plan.

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Neutronics Assessment of Molten Salt Breeding Blanket Design Options

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Molten salts have been considered as a breeding material and coolant candidates in fusion systems. Flibe consisting of LiF and BeF_2 with mole ratio of 2:1 has been widely considered. It has the attractive features of low activation, low chemical reactivity with air and water, low electrical conductivity, and good neutron attenuation properties. On the other hand, it has a relatively high melting point (459°C), low thermal conductivity, tritium permeation concern, and requires control of the corrosive TF and F_2. A low melting point Flibe (380°C) was also considered but it has much higher viscosity. The molten salt Flinabe that consists of LiF, BeF_2 and NaF has recently been considered due to its low melting point (305°C) and vapor pressure. The breeding capability of the molten salt is limited, requiring separate neutron multiplier.

A study is underway to identify attractive molten salt breeding blanket concepts that can be utilized in fusion power plants. Special attention is given to concepts that can be developed, qualified and tested in the time frame of ITER. For this reason, the conventional ferritic steel alloy F82H with a temperature limit of 550°C is considered as the structural material. Three molten salt blanket concepts were evaluated. The first concept is a self-cooled Flinabe blanket with Be multiplier (SC). It uses an innovative re-circulating coolant scheme, which allows effective cooling of the first wall (FW) while enhancing the outlet temperature. The other concepts are dual coolant options with helium cooling the FW and blanket structure, Flibe breeder, and either Be (DC-Be) or lead (DC-Pb) as neutron multiplier. If the high melting point Flibe is used in the DC concepts, the FW should be plated with ODS ferritic steel to allow higher temperatures. Otherwise, the low melting point Flibe or Flinabe should be used.

In this paper, the neutronics assessment of the three concepts is presented. The ARIES-RS configuration with a peak neutron wall loading of 5.45 MW/m^2 was used. Several iterations were made to determine the blanket radial build that achieves adequate tritium breeding ratio (TBR). Larger margins were considered to account for uncertainties resulting from approximations in modeling. The same TBR can be achieved with a thinner SC OB blanket (50 cm) compared to the DC blanket (65 cm). A thicker Be zone is required in designs with Flinabe. The overall TBR will be ~1.17 excluding breeding in the divertor region. We conclude that the three design concepts have the potential for achieving tritium self-sufficiency. Several design parameters can be adjusted (e.g., multiplier thickness and blanket thickness) to ensure tritium self-sufficiency if necessary. Using Be yields higher blanket energy multiplication (1.27 for SC, 1.21 for DC-Be, and 1.13 for DC-Pb). Modest amount of tritium is produced in the Be (<3 kg) over the blanket lifetime of ~2.4 FPY. The tritium inventory will be much smaller depending on the Be temperature. Using He gas in the DC blanket results in about a factor of 2 lower blanket shielding effectiveness. With a total blanket/shield/VV radial build of 105 cm in the IB and 120 cm in the OB it is possible to ensure that the shield is a lifetime component, the VV is reweldable, and the magnets are adequately shielded. Based on this analysis we conclude that molten salt blankets can be designed for fusion power plants with neutronics requirements such as adequate tritium breeding and shielding being satisfied.

Innovative Liquid Blanket Design Activities in Japan

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After selecting and proposing the molten-salt Flibe blanket in the LHD-type helical reactor FFHR in 1995 [1], intensive design studies on Flibe blanket have been continued and expanded into wide R&D areas, including materials compatibility, tritium disengager system, advanced thermofluid and MHD effects, and heat-exchanger system, in the NIFS collaborations under the Fusion Research Network in Japan Universities with international research programs such as JUPITER-II. The main purposes of these activities are to clarify key engineering issues and to enhance key R&D activities required for advanced system integration of D-T power plants. Recently an improved long-life Flibe blanket has been proposed, and the self-cooled Li/V blanket design has started, based on intensive R&D works and results. This paper mainly focuses on these liquid blanket designs and related R&D activities in Japan.

New design approaches are proposed for the Flibe / RAFS (reduced activation ferritic steel) blanket in FFHR to solve the major issues of blanket space limitation and replacement difficulty. For the neutron wall loading of $1.5MW/m^2$ as adopted so far in FFHR designs, an innovative concept of replacement-free blankets is possible in the reactor life of 30 years, using carbon armor tiles as ISSEC (Internal Spectral Shifter and Energy Converter), in which the tritium breeding and tiles cooling were key issues [2]. In our new design of the Spectral-shifter and Tritium breeder Blanket (STB) of Flibe in the limited thickness of about 1m, where the Be layers in the both C and Flibe zones are optimized, the fast neutron flux at the first wall of RAFS is reduced to about 1/3 the original flux. At the same time the local TBR of 1.3 is kept, and the fast neutron fluence to the super-conducting magnets is reduced to $8x10^{22}$ n/m^2. According to a preliminary thermal analysis, the surface temperature of the carbon armor is about 1,600°C under conditions of an effective thermal conductivity of 100W/m/K for C-Be-C bonded armors and heat transfer coefficient of $6,000W/m^2/K$ for the mechanical contact using a graphite sheet.

As for heat transfer issues in Flibe, the recent R&D results in the TNT (Tohoku-NIFS Thermofluid) loop experiments using HTS (Heat Transfer Salt) are quite promising. In fact it is clarified that the heat transfer enhancement using packed-bed tube is superior to that of turbulent heat transfer under the same flow-rate condition [3]. The key R&D issues to develop the STB concept, such as radiation effects on carbon and enhanced heat transfer of Flibe, are elucidated.

From the aspects of high thermal conductivity and tritium self-sufficiency without neutron multiplier Be, a liquid Li blanket is another attractive concept. As for MHD pressure drop, the relation between coating material electro-conductivity and allowable crack fraction has been made clear using 2D modeling [4]. Encouraged with intensive R&D progress in MHD coating and high purity V fabrication, optimization studies of Li/V blanket designs are in progress [5].

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Ceramic Breeder Blanket for ARIES-CS

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The first phase of the ARIES-CS study has focused on scoping out maintenance schemes and blanket designs best suited for a compact stellarator configuration. The study will then down-select to a couple of most attractive combinations of blanket configuration and maintenance scheme for more detailed studies culminating in the choice of a point design for a full system design study. One of the blankets developed during the early scoping phase is a helium-cooled ceramic breeder blanket. Consistent with the guidelines of the study, this concept was developed to an extent sufficient for a credible case to be made regarding performance, fabrication and maintenance.

Ceramic breeder designs tend to favor a modular configuration which, in the case of a compact stellarator, provides the flexibility in setting module sizes best suited to the particular reactor geometry. This also assumes a modular maintenance approach through ports. A ceramic breeder design has traditionally been coupled to a Rankine steam cycle since the maximum temperature of the coolant tends to be limited by the maximum temperature limit of the structural material (typically ferritic steel, FS). However, safety concerns have been raised about the possibility of a tube rupture in the module followed by a tube rupture in the steam generator which could eventually result in unacceptable steam/Be interaction in the case of failure of the pressurized module. Thus, unless a clear case could be made that such an accident is beyond design basis, the module would need to be designed to accommodate the pressurization, which translates into more structure and less tritium breeding. To avoid this issue, it was decided to reconsider the possibility of coupling a Brayton cycle to such a blanket, by optimizing the cycle as well as by maximizing the coolant temperature through limited utilization of ODS FS in high temperature regions, for example as an outside first wall layer diffusion bonded on regular FS.

The blanket module is then designed to accommodate a relatively low pressure of about 0.5-1 MPa compared to a He pressure of about 8 MPa. The overall configuration is relatively simple, consisting of a number of breeder and Be multiplier packed bed layers separated by cooling plates and arranged in parallel to the first wall. The dimensions are optimized to meet the breeding requirement and to accommodate the maximum temperature limits of the different materials. As fabrication technique, it is envisaged to fill the breeder regions outside the canister utilizing a membrane to contain the breeder pebbles, and then to insert the breeder unit in the canister which is finally filled with Be pebble beds.

This paper describes the conceptual design of this ceramic breeder blanket. Key parameters are summarized and major issues are discussed.

Oral Session II-6

IFE Designs and Technology

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Development Path for Z-Pinch IFE

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The goal of z-pinch IFE is to extend the outstanding single-shot z-pinch ICF results on Z to a repetitive-shot z-pinch power plant concept for energy. On Z, the high magnetic field pressures associated with 20-MA load currents implode a wire array z-pinch, generating up to 1.8 MJ of x-rays at powers as high as 230 TW. Using a *double-pinch hohlraum target*, capsule implosions in the ~70 eV hohlraum have been radiographed by 6.7 keV x-rays produced by the Z-Beamlet Laser (ZBL). These experiments demonstrated capsule implosion convergence ratios between 14 and 21 from a radiation drive symmetry that is within 1.6 to 4 times the symmetry required for scaling to high yield. Using a *dynamic hohlraum target*, a 2.1-mm-diameter deuterium-filled CH capsules absorbs up to 35 kJ of x-ray energy from the ~ 220 eV dynamic hohlraum. The capsule convergence ratio is 5-10 and the thermonuclear DD neutron yield is up to 8 $\times 10^{10}$. These yields approach being a factor of 10 higher than that achieved by any other indirect-drive target experiments.

Based on (1) these demonstrated z-pinch driven target results, (2) the high demonstrated electrical conversion efficiency (~ 15%) on Z from wall-plug to x-rays, and (3) the lowest cost in $ /Joule for all IFE drivers, it appears that z-pinches are particularly attractive for IFE provided a suitable method for rep-rated standoff (separation of driver and target) can be developed. The simplest and most robust method is the Recyclable Transmission Line (RTL) concept. In this concept, an RTL is made from a solid coolant (e.g., Flibe), or a material that is easily separable from the coolant (e.g., low activation ferritic steel). The RTL/target assembly is inserted through a single opening at the top of the thick liquid wall power plant chamber. The shot is fired, portions of the RTL are vaporized and end up mixed with the coolant to be recycled, the upper remnant of the RTL is removed, and the cycle is repeated. The present strategy for Z-Pinch IFE is to use high yield (~3 GJ/shot) and low repetition rate per chamber (~0.1 Hz).

A development path for Z-Pinch IFE has been created that complements and leverages the NNSA DP ICF program. This path includes a Proof-of Principle (PoP) phase, an Integrated Research Experiment (IRE) phase, an Engineering Test Facility (ETF) phase, and a Demo phase. Funding by a U.S. Congressional initiative of $ 4M for FY04 through NNSA DP is supporting assessment and initial research on (1) RTLs, (2) repetitive pulsed power drivers, (3) shock mitigation [because of the high yield targets], (4) planning for a proof-of-principle full RTL cycle demonstration [with a 1 MA, 1 MV, 100 ns, 0.1 Hz driver], (5) IFE target studies for multi-GJ yield targets, and (6) z-pinch IFE power plant engineering and technology development. Initial results from all areas of this research will be discussed.

*In collaboration with G. E. Rochau, S. A. Slutz, T. A. Mehlhorn, M. K. Matzen, J. P. Quintenz, W. B. Gauster, and the Z-Pinch IFE Team.

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Update on Progress and Challenges in the Development of Heavy Ion Fusion

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Since the completion of the Robust Point Design for heavy ion fusion power plant, progress has been made in addressing key issues and some possible new directions have emerged. Work has continued on thick liquid wall chambers, which are particularly well suited to heavy ion fusion since they allow compact chambers minimizing the stand-off for the final focus magnets and thus improving beam focusing on target. University experiments have continued to address issues related to the disruption and re-establishment of the protective liquid layer in a rep-rated system, including repetitive disruption of jet arrays at UC Berkeley. ARIES-IFE study of thick liquid chambers contributed to our understanding of possible design constraints due to drop and aerosol formation. Studies at Georgia Tech demonstrated the need for flow conditioning and boundary layer trimming of jets in order to minimize excessive drop formation.

Work has continued on HIF indirect-drive targets in an attempt to allow larger spot sizes than the ~2 mm spots required by the baseline target in the RPD. A promising approach is the use of shine shields at both ends of the hohlraum, which allows beams to fill the entire 5-mm-radius target. Radiation flow around the shine shields results in hot spots on the capsule (P_4), but it has been proposed to handle these with shims on the capsule. Experiments on the effectiveness of shimmed targets are being conducted on Sandia's Z-machine.

This larger spot size target design opens the door to other driver and focusing schemes. Work has started exploring a modular driver approach in which many individual accelerators provide the total beam energy on target as apposed to the single accelerator with many (~100) individual beamlets threading common induction cores. The currently favored approach is to use much lower mass ions (e.g., Ne with A = 20 amu) and solendoid magnets in the accelerator instead since solenoids allow transport of very high current beams. Hybrid designs, in which the first half or more of the accelerator uses solenoids and the rest uses quadrupoles (since quad focusing is more efficient at high ion kinetic energy) are also being investigated. Neutralized drift compression and plasma channel focusing are being studied as a way to deliver these high current beams to target. We have also started investigating new chamber designs that would be compatible with this focusing scheme, in particular designs using a vortex flow configuration to establish the thick-liquid-wall.

This talk highlights progress since the last Technology of Fusion Energy Conference in these areas and points out the challenging next steps in the R&D to develop HIF power plants.

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Overview of Magneto-Inertial Fusion

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Magneto-inertial fusion (MIF) is a pulsed high energy density approach to achieving fusion in that combines features of both inertial (ICF) and magnetic fusion (MFE) techniques. When a magnetic field is embedded in a warm dense plasma, thermal insulation is improved, thereby allowing compression to be achieved with the use of lower power (and hence cheaper) inertial drivers. At the ICF end of the spectrum, one might consider ICF targets with an embedded external magnetic field. At the other end of the spectrum, we consider “fast” adiabatic compression of conventional magnetic fusion plasmas. Magnetized Target Fusion (MTF) starts with a modest magnetic field in a warm ~100 eV plasma, and then radially compresses it with a metallic flux conserver (liner) by factors of 10, to achieve fusing plasma conditions. At Los Alamos we are developing a plasma for magnetized target fusion experiments, consisting of a small field reversed configuration (FRC) plasma, called FRX-L, which will be translated into a region surrounded by an aluminum liner. Later (in the next few years) we plan to demonstrate the physics of plasma/liner implosions for MTF jointly with the Air Force Research Laboratory, at the Shiva Star pulsed power driver facility in Albuquerque. This talk will cover issues relevant to MIF, including present research, ranges of expected fusion gain, batch burn, fusing versus burning plasmas, possible reactor concerns such as driver standoff, liquid-walled chambers, and debris from the liner. This work is supported by the DOE Office of Science, and contract #W-7405-ENG-36.

Present Status of Fast Ignition Research and Prospects of FIREX Project

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We have been studying the fast ignition scheme of laser fusion by the GEKKO XII and PW lasers. In the 2002 IAEA FEC, it was reported that the neutron yield increased from 10^4 without heating to 10^7, when a 400 J/0.6 ps PW laser was injected into a compressed CD shell. This indicates that the core plasma temperature increases by 500 eV and the energy coupling efficiency between heating laser and core plasma is 20-25% [1]. According to these results, we started the FIREX (Fast Ignition Realization Experiment) project toward demonstrating the fast ignition with a new high energy PW laser, LFEX (Laser for Fast Ignition Experiment) which is currently under construction. In this paper, the progresses in the experimental research on scientific and technological issues related to fast ignition and the integrated code development toward the FIREX will be reported.

The new heating laser LFEX has been designed to deliver 10kJ energy in 10ps with 1ps rise time. It consists of 2x2 segment amplifiers with a size of 40 cm x 40 cm in each beam. After pulse compression, 4 laser beams are combined into 1 beam for obtaining a single focal spot without reducing the encircled energy. In the pulse compressor, the segmented gratings will be used because of the limited size of the dielectric grating. In the preliminary experiment by Ti:sapphire laser no significant changes in compressed pulse width and far-field pattern were observed between the segmented gratings and a monolithic grating. One of the critical technical issues of the LFEX is the coherent combining of 4 segmented beams. A precise phase control system in beam and among beams is introduced. The LFEX is under construction and the FIREX-I experiment will start before 2007 .

In the fast ignition scheme using cone-shell target, it is important to understand the physics of ultra-intense laser absorption, relativistic electron generation and transport in the cone and the compressed plasma, as well as non-spherical cone-shell target implosion. As interpretation of the cone-shell-target experiment, the relativistic electron propagation in dense plasmas has been widely investigated by experiment, simulation and theory. Further analysis of the cone target experiments has been carried out by the integrated simulation combining PIC, 2D hydro PINOCO and Fokker Planck (F.P.) codes[2]. We found that the unique interaction of heating laser with cone contributes to enhancing the coupling efficiency of short pulse laser to core plasmas. Namely, the energy distribution of relativistic electron has a low energy component, which contributes to the efficient heating. The non-spherical cone-shell target implosion was investigated by experiments with the GEKKO XII laser[3]. The experiments indicate that the imploded plasma ρr is not smaller than that of the corresponding spherical CD shell implosion.

The heated plasma parameters and the gain for the FIREX-I were evaluated to be ρr = 0.15g/cm^2, = 20 μm, T = 8 keV, and Q = 0.1 by the integrated simulation with 0.5 MeV slope temperature for the 10kJ heating pulse energy. When the above expected plasma parameters are achieved, we plan to start the FIREX-II project in which both implosion and heating lasers are up-grated to 50 kJ. In this case, the gain will reach higher than unity and the ignition will be achieved. Toward the FIREX-I experiment, we also started research and development for fabricating cryogenic foam shell cone target as the collaboration work with NIFS (National Institute of Fusion Science). A cryogenic target will be imploded in a year.

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The Modular Approach to Heavy Ion Fusion

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We report on an ongoing study on modular Heavy Ion Fusion drivers. The modular driver is characterized by 10 to 20 nearly identical induction linacs, each carrying a single high current beam. In this scheme, the IRE can be one of the full size induction linacs. Hence, this approach offers significant advantages in terms of driver development path. For beam transport, these modules use solenoids, which are capable of carrying high line charge densities, even at low energies. A new injector concept allows compression of the beam to high line densities right at the source. The final drift compression is performed in a plasma, in which the large repulsive space charge effects are neutralized. Finally, the beam is transversely compressed onto the target, using external solenoids or current-carrying channels, in the Assisted Pinch Mode of beam propagation. We will report on progress towards a self-consistent point design from injector to target. Considerations of driver architecture, chamber environment as well as the methodology for meeting target requirements of spot size, pulse shape and symmetry will also be described. Finally, some near-term experiments to address the key scientific issues will be discussed.

Dynamics of Liquid-Protected Fusion Chambers

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Thick liquid jets can provide vital protection for inertial fusion power plants by shielding solid structures inside the reaction chamber from high-energy neutrons and x-rays. Fluid mechanics problems then replace the complex and costly materials testing and development path required to develop alloys capable of surviving an unattenuated neutron spectrum.

At U.C. Berkeley, thick liquid protection is being studied for several inertial fusion energy (IFE) concepts via numerical and experimental tools. TSUNAMI, a gas dynamic code, is being used to model the venting of debris and vaporized liquid following fusion micro-explosions in various thick-liquid protected chambers. These simulations set the initial conditions for the liquid response studies that are investigated experimentally.

A scaled impulse delivery system able to deliver precise loads is being implemented to study the response of target facing jets to the forces generated by fusion reactions in both thick-liquid protected Z-Pinch and heavy-ion fusion chambers. Those experiments are conducted in a sealed vacuum container called the Vacuum Hydraulics EXperiment, VHEX.

The thickness and surface smoothness of vortex flows needed to protect heavy-ion beam tubes in these latter chambers have been investigated optically. A mineral oil that allows a close match of the molten salt flibe dynamic similarity has been identified and used for these experiments.

The development of new larger vortex flows, which could potentially be used with solenoid focused heavy-ion beams and even magnetic Fusion Energy applications, is underway. The possibility to obtain and control a stable layer is being investigated.

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Oral Session III-1

Materials Development

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Overview of the US Fusion Materials Sciences Program

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This presentation will provide an overview of recent experimental and modeling research highlights on advanced structural materials for fusion energy systems. A series of high-performance structural materials have been developed over the past ten years with significantly improved properties compared to earlier materials. Recent advances in the development of high-performance ferritic/martensitic and bainitic steels, nanocomposited oxide dispersion strengthened ferritic steels, high-strength V alloys, improved-ductility Mo alloys, and radiation-resistant SiC composites will be reviewed. Multiscale modeling activities are providing important insight into defect production and migration mechanisms, plastic deformation mechanisms, and fracture mechanics behavior. In-situ straining experiments performed in electron microscopes have provided new information on the deformation processes that occur in irradiated metals. Fundamental modeling and experimental studies are in progress to determine the migration and trapping behavior of helium in metals, in order to design materials with improved resistance to void swelling and high temperature helium embrittlement in fusion reactor irradiation environments. Recent chemical compatibility tests have identified several promising new candidates for magnetohydrodynamic (MHD) insulators in lithium-cooled systems, and have determined that SiC has good chemical compatibility with Pb-Li up to very high temperatures. Work on advanced joining techniques such as friction stir welding will also be described. Work performed by US materials science researchers in support of the ITER international team activities will be briefly described. Finally, research in progress as part of a US-Japan collaboration to investigate the effects of fusion-relevant helium generation rates on the mechanical properties and microstructures of neutron irradiated advanced ferritic/martensitic steels will be summarized.

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European Fusion Materials Research Program - Recent Results and Future Strategy

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The European Long Term program is focused on the R&D of the materials and key technologies needed for a demonstration (DEMO) reactor, and in particular, on structural materials for the blanket and the divertor. The strategy is based on the assumption of a ‘fast track’ approach: (i) ITER operation staring in 2015, (ii) IFMIF first beam operation (with half power) in 2017, (iii) DEMO design starting around 2022. In the European power plant conceptual studies (PPCS) several types of breeding blankets and divertors have been considered utilizing moderate extrapolation up to very advanced concepts, in both, physics and technology. This planning determines the time schedule for the materials development.

In the EU approach DEMO relevant Test Blanket Modules (TBMs) are to be installed in ITER from day one of operation. Therefore, fully developed and code qualified technologies and materials (up to 3dpa) have to be ready by 2015. Two different breeding options, i.e. a helium cooled lithium lead and a helium cooled ceramic blanket with pebble beds are considered for early DEMO operation. The reduced activation ferritic/martensitic (RAFM) steel, called EUROFER, is the candidate structural material designed to operate at 250-550°C.

In parallel to the development of ‘generation one’ breeding blankets and materials, research on alternative, so-called ‘advanced’, materials has to be conducted for operation at higher temperature. In the long term the European Union therefore sees a logical sequence, by supplementing EUROFER with the next generation of advanced ferritic steels based on oxide dispersion strengthening (ODS), and of fibre-reinforced Silicon Carbide (SiC_f/SiC).

RAFM steels of the EUROFER family will undergo several steps of development within the next two decades: from the early use in ITER TBM to a material qualified up to neutron doses of 150 dpa for DEMO application in about 2022. The strategy foresees that, after optimization of mechanical and irradiation properties, in the following step the ‘low level waste’ standard will be achieved.

The development of ODS steels, started in 2002, is also aiming to be fully developed by 2022 with an operational window that is shifted about 100°C towards higher temperatures. SiC/SiC composites will be used as functional materials inserts in steel blanket structure during the early stage of a DEMO operation. In about 30 years from now, fully qualified SiC/SiC should be available as structural material for a second generation of DEMO blankets and for power plant applications.

Tungsten alloys are considered as structural material for high temperature applications in Gas cooled divertors in DEMO and beyond. They show promising physical properties, but they also exhibit low fracture toughness and suffer from irradiation embrittlement even at elevated temperatures of 600°C and above. In addition, they have, so far, not been used as structural materials for larger components. Therefore, the development of W alloys to be used from 600-1200/1300°C is a real challenge and needs a coordinated international effort.

Recent accomplishments and future prospects of materials R & D in Japan

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After more than two decades of the intensive research activities on fusion engineering at Universities and JAERI, these activities have been unified under the management of the ministry of education, culture, sports, science and technology (MEXT) with the emphasis on fusion materials and blanket engineering.

To meet the definition of fusion reactor materials R & D, three categories of structural materials have been studied, those are; reduced activation martensitic/ferritic steels (RAFs), vanadium alloys and SiC/SiC composite materials. The R & D histories of these candidate materials and the present status will be reviewed with the emphasis on materials behavior under radiation damage.

The present status of RAFs is the most matured and the representing RAFs for fusion application, such as F82H and JLF-1, have been studied even under the IEA working group activities for more than ten years and are almost ready to be used to near term plasma devices or ITER test modules.

Significant progress has been made recently in fabrication and welding technology, applicable to industrial scale manufacturing, for V-4Cr-4Ti alloys by improved control of interstitial impurities. Recent efforts are focused on MHD insulator coating development as a key issue for vanadium alloy/liquid lithium blanket. Development of advanced vanadium alloys by minor addition of Y, Al and Si is also in progress for improved radiation and oxidation resistance.

The materials R & D methodology has been quite unique for the case of SiC/SiC and the typical example of the process development is emphasized. Such as, CVI methods and NITE process, where near near stoichiometry fibers and matrices were employed to provide excellent total performance. The recent extensive activities trying to make real size reactor components will be also presented. These results are encouraging to make attractive fusion reactors, satisfies 3E requirement, utilizing SiC/SiC composites as major structural materials.

Based on the progresses in structural materials R & D, JUPITER-II, the phase-4 of the Japan/USA collaboration on fusion materials research, has been initiated since April 1, 2001. This is the integration activity of blanket and materials engineering, where self cooled liquid blanket and He-gas cooled solid blanket systems are of concern. The brief introduction of JUPITER-II program, together with the JAERI/DOE Phase-4 collaboration on neutron radiation effect study on RAFs, will be presented.

New Superconductors for Fusion Magnets

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All fusion magnets until now have been made from Nb-Ti, Nb_3Sn or occasionally Nb_3Al. In 2003 several very important milestone in superconducting magnet technology occurred – a 16 Tesla dipole using advanced, very high current density Nb_3Sn was made at LBNL and a 25 Tesla small bore Bi-2212 solenoid was made at the NHMFL. Superconducting materials made major advances too. Bi-2223, the only industrially available cuprate high temperature superconductor, was optimized to new high values of current density, while coated conductors of YBCO were first made by multiple continuous processes, leaving hope that cheap, Ag-free conductors will be soon industrially available. In 2003 too it was shown that MgB_2 could be alloyed to produce critical fields of over 40 Tesla, thus convincingly exceeding the critical field properties of any Nb-base superconductor. This talk will address the future technology promise of new superconductors, taking advanced Nb_3Sn, the present choice for fusion magnets, as their benchmark.

Status of Tritium Permeation Barrier Development in the EU

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Tritium permeation can be significantly reduced by a suitable barrier on the structural materials of a future fusion power plant. Since alumina has the capability of tritium permeation reduction, the development of such coatings on ferritic martensitic steels by different techniques like hot-dip aluminising (FZK), vacuum plasma spraying (JRC Ispra) and chemical vapour deposition (CEA) was funded by the EU during the last 10 years. The final objective was to identify a so-called reference coating for structural components of the WCLL blanket.

The presentation describes the process specifica and the results of the corresponding hydrogen permeation measurements, performed at ENEA, Brasimone, Italy. Only the results for CVD and HDA coating show clearly, that the required PRF values of > 1000 in H_2 gas were sufficiently exceeded, but lower values were obtained in the Pb-17Li environment. The post mortem analysis showed that surface defects and spallation of parts of the coatings were responsible for the low PRF's. The reason for this behavior is not fully clear at the moment and needs additional investigations. Funding of the EU regarding the R&D activities was postponed in 2002, until finishing of the evaluations of further needs for tritium permeation barriers after the redesign of the european blanket concepts.

Recent Progress Addressing Compatibility Issues Relevant to Fusion Environments

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There are numerous potential compatibility issues in fusion reactors. Specific concerns are determined by reactor concept, including the cooling and tritium breeding schemes. This paper addresses research aimed at understanding critical scientific and technological factors associated with several key areas of current interest.

The greatest effort is devoted to examining the compatibility of ceramic materials with molten Li. Electrically insulating coatings are needed on the first wall of magnetic confinement reactors to reduce the pressure drop due to the magneto-hydrodynamic (MHD) force on the flowing Li. Very few candidates are sufficiently compatible with Li at high temperatures (600-800°C), but the most promising candidates are AlN, Y_2O_3 and Er_2O_3. Coatings of Y_2O_3 have been marginally successful in static Li capsule tests and Er_2O_3 coatings are currently being investigated. Cracks cannot be tolerated in this coating because Li will likely wet the crack surfaces. Therefore, an outer vanadium layer will likely be required . This outer layer also may significantly inhibit the solid state reaction between the ceramic coating and Li. In-situ experiments are being developed to test this dual-layer concept.

While SiC composites are being widely investigated because of their high temperature strength, limited information is available about its compatibility with Pb-Li at temperatures of 800°C and higher. As a first step in assessing the maximum operating temperature for this concept, specimens of dense, high purity monolithic SiC were exposed to Pb-17at.%Li for 1000h in isothermal capsule tests at 800° and 1100°C. After 1000h at 800°C, no evidence of reaction or wetting between SiC and Pb-17Li was observed. No specimen mass change was observed after 1000h at 1100°C, but there was evidence of limited wetting. These results suggest that SiC is compatible with Pb-17Li to at least 1100°C in a static environment.

Finally, some work on the oxygen uptake kinetics in vanadium alloys and the effect on mechanical properties continues. Prior work in low oxygen pressures found linear kinetics at the lowest pressures relevant to a high purity fusion environment and an appropriate model was developed. The current effort is examining kinetics at 700°C in high purity He environments to determine the effect of system pressure and impurity content.

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Oral Session III-2

High Heat Flux Components

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"Overview of the ALPS Program"[1]

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The US Advanced Limiter-divertor Plasma-facing Systems (ALPS) program is working to develop the science of liquid metal coated divertors for near and long term tokamaks. These systems may help solve the very demanding heat removal, particle removal, and erosion issues of fusion plasma/surface interactions. We are designing both static and flowing liquid lithium divertors for the National Spherical Torus Experiment (NSTX) at Princeton. We are also studying lithium, tin, and gallium divertors for future D-T machines, with emphasis on key science/engineering issues.

ALPS combines tokamak experiments, lab experiments, and modeling. Lab experiments PISCES(UCSD), IIAX(UIUC), ARIES(SNL), PRIME(ANL) show strong deuterium pumping by liquid lithium, enhanced but stable D and self-sputter yields up to surface temperatures of ~450°C, and are providing data on lithium on graphite substrate performance. CDX-U(PPPL) tokamak experiments show good plasma performance using a full liquid lithium limiter. DIII-D (GA) experiments with a small DiMES probe have provided key data on Li sputtering, transport, and disruption/operational limits.

Molecular dynamic modeling of D and He transport in liquid metals is being used to understand diffusion, trapping, and temperature enhanced sputtering. There is some indication that lithium may be able to pump helium via trapped bubbles—if true, an important advantage for future D-T device application.

The two proposed systems for NSTX are a static pre-shot deposited ~300 nm liquid Li divertor coating, and a 10 m/s in-shot injected flowing system. Both systems are predicted—via detailed coupled code/data analysis—to strongly pump D^+ resulting in a low-recycle high plasma temperature (~200-400 eV), low density, edge/SOL regime, with potentially major advantages for the NSTX physics mission. The static system has acceptable net sputter erosion and low core plasma contamination, and has little or no MHD issues. The more complex flowing system would be even more capable (higher power handling) but needs critical MHD evaluation, currently underway via experiments LIMITS(SNL), MTOR(UCLA), and code work.

Modeling of reactor grade systems indicates that tin and possibly gallium coated divertors would operate in a "conventional" high recycle regime with good power handling capability and very low sputter erosion/plasma-contamination. Whereas evaporation/sheath-superheat analysis shows acceptable Li operation up to about 500 °C, much higher limits, ~1200 °C, obtain for Sn and Ga, however sputter yield increase with surface temperature may impose lower limits. (In general, the strong plasma flow to the *divertor* substantially eases evaporation and other concerns, compared to e.g., a liquid *first wall* system). Supporting lab experiments are being conducted on critical issues of temperature-dependent particle/surface interactions in liquid tin and gallium.

Erosion due to ELMs and other transients is being examined with the HEIGHTS code package. There is erosion/contamination concern for all materials examined, solid and liquid, but with the obvious advantage for the liquids of being able to replenish the surface via flow.

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[1] Work supported by the US Dept. of Energy

EU Development of High Heat Flux Components

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The development of plasma facing components for next step fusion devices in Europe is strongly focused to ITER. From a heat flux point of view, the divertor represents the most critical plasma facing component which has to withstand 3000 cycles at a power density level up to about 3 MWm^{-2} for the upper part of the vertical target which will be manufactured from flat W-tiles or W-monoblocks attached to a CuCrZr-heat sink. The lower straight part of the divertor will be subjected to even higher thermal loads (up to 20 MWm^{-2} for about 10% of plasma discharges during the so-called 'slow transients'). The reference design for this component is based on the monoblock principle with 3-directional CFC tiles attached to a straight coolant tube made from a precipitation hardened copper alloy with twisted tape inserts.

For the primary first wall panels of the shielding blanket modules, the EU has developed a design based on a key system mechanically attached to the front of the shield. Here the plasma facing panels consist of beryllium tiles which are joined by brazing or by Hot Isostatic Pressing to a water cooled bi-metallic structure made from a CuCrZr or CuAl25 alloy heat sink layer and a 316L(N) stainless steel back plate. These components shall be designed for a peak heat flux of 0.5 MWm^{-2} for up to 30,000 cycles.

Electron beam simulation experiments have been used to investigate the performance of high heat flux components under ITER specific thermal loads. These tests have been performed on a wide spectrum of different design options, including the above mentioned reference solutions for ITER with tungsten, CFC, and beryllium armor. Beside thermal fatigue tests which are primarily focused to the integrity of the joint between the plasma facing armor and the heat sink, also transient events which occur during off-normal plasma operation with deposited energy densities up to several tens MJm^{-2} have been investigated experimentally. These events are expected to occur on a time scale of a few milliseconds (plasma disruptions) or several hundred milliseconds (vertical displacement events) and have been identified as a major source for the production of neutron activated metallic or tritium enriched carbon dust which are of serious importance from a safety point of view.

The irradiation induced material degradation is another critical concern for future D-T-burning fusion devices. In ITER the integrated neutron fluence to the first wall and the divertor will remain in the order of 1 dpa and 0.2 dpa, respectively. This value is low compared to future commercial fusion reactors; nevertheless, a non-negligible degradation of the materials has been detected, both for mechanical and thermal properties, in particular for the thermal conductivity of carbon based materials. Beside the degradation of individual material properties, the high heat flux performance of actively cooled plasma facing components has been investigated under ITER specific thermal and neutron loads.

Plasma/Liquid-Metal Interactions during Tokamak Operation

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One of the critical technological challenges of future tokamak fusion devices is the ability for plasma-facing components to handle both normal and abnormal plasma/surface interaction events that compromise their lifetime and operation of the machine.

During normal operation of H-mode, edge-localized modes (ELMs) are a serious concern for divertor and nearby plasma-facing components (PFCs). During ELMs part of the total plasma energy is released and deposited on the divertor surface during 0.1 to 1.0 ms with a frequency of between 1 to 20 Hz, depending on the ELM type. The power from the scrape-off-layer (SOL) to the PFC in ITER-like devices can then increase from 5 MW/m^2 to ≈ 300-3000 MW/m^2. Erosion lifetime strongly depends on ELM power deposited on PFC. Moreover, the resulting evaporated material can reach the core and disrupt the plasma. In addition, with higher ELM frequency, thermal cycling takes place and can result in thermal stresses and fatigue. At high ELM power, the resulting high surface temperature causes vapor cloud formation with similar consequences to disruptions. Vapor shielding decreases energy deposition at the surface but increases radiation flux to nearby components. Metallic PFC can melt and liquid metal flow instabilities occur with mass losses due to both MHD splashing effects and vaporization. A comprehensive two-fluid model is developed to integrate Core and SOL parameters during ELMs with PFC surface evolution using HEIGHTS package.

In addition to thermal erosion due to ELMs, physical erosion (i.e., sputtering) can also be enhanced. The significant increase in particle flux to the divertor and nearby PFCs can enhance sputtering erosion by an order of magnitude or more. Advanced designs of first wall and divertor systems propose the application of liquid-metals as an alternate PFC to contend with high-heat flux constraints of large-scale tokamak devices. Liquid-metals considered include: lithium, tin, gallium, flibe, and tin-lithium. However, replenishable liquid surfaces also have concerns. Enhanced liquid erosion with temperature and with incident particle energy has been measured for most of these candidate PFC materials. Monte Carlo and Molecular Dynamic methods along with experimental data are used to calculate the enhanced erosion rates.

Initial results indicate that high-power ELMs in ITER-like machines can cause serious damage to PFCs, may terminate plasma in disruptions, and because of large contamination may affect subsequent plasma operations. Erosion lifetime and plasma contamination for solid and liquid PFCs are studied in ITER-like devices.

IFE First Wall Survival : Development and Testing of a Refractory Armored Ferritic

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Two unique characteristics driving IFE first wall design are the intense pulsed heat loading and the significant levels of high-energy implanted helium. A tungsten coating on ferritic steel is being developed to handle the spiked heating of IFE while retaining the benefit of low activation. This paper will present the development and testing of refractory armored ferritics. Specifically, various fabrication methods were explored including the primary candidate vapor plasma spraying. Testing has focused on the response of the tungsten coating to helium implantation and cyclic thermal heating. Results indicate that the high-dose helium implantation, which was a serious concern to cause spalling of the tungsten surface leading to unacceptable erosion, is likely manageable through prudent selection of tungsten microstructure, chamber size, and annealing temperature. The second major area of testing has been on the thermal fatigue of the tungsten/ferritic duplex using a pulsed infrared testing stand. Specifically, tungsten/ferritic samples have been thermally fatigued to 10,000 pulses at a peak of 23.5 MW/m^2 without interfacial failure. In addition to a review of this testing and development work, next generation “engineered” structures under studied will be presented.

The Use of Ga and Li as Limiter Materials in T-3M and T-11M Tokamaks

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The limiter and divertor plates are most critical tokamak plasma facing components (PFC) with highest heat load. This load has character of short periodical pulses with melting and evaporation of PFC during plasma disruptions and Edge Local Mode development. As a best candidate materials for PFC should be a liquid metals (LM), which admit a permanent PFC renovation. This advantage of LM is known since from UWMAK-Project (1974), but the practical use of liquid metals has some potential obstacles in real tokamaks. In particular:

- a liquid metal splashing under MHD-forces in plasma instabilities,
- a probably abnormal liquid metal erosion, as result of: ion sputtering, unipolar arcs and instabilities of liquid metal –plasma boundary,
- a technological problems of liquid metal injection in strong magnetic fields.

The attempts to overcome this obstacles were undertaken in Russian T-3M and T-11M tokamaks ($I_p \leq$ 100kA $B_T \approx$1T) on 1990-2004 Years.

In T-3M droplet stream and film flow Ga limiters were tested. In T-11M were performed the experiments with Li Capillary Pore Systems (CPS) as rail limiter for modeling of real Li erosion in typical tokamak boundary condition (T_e = 30 ±10 eV, P_{load}=10 MW/m^2, D, He plasma).

The main results are:

1. The Ga-droplet stream can be useful as method of LM injection in tokamak. The Ga-film limiter experiment was unsuccessful.
2. The Li-CPS limiter experiment in T-11M was successful. It was received the quasi state (0.2s) clean deuterium plasma with Z_{eff}=1.1±0.1 (T_e(0)= 0.4keV) and plasma boundary cooling by Li-radiation.
3. The Li emission from CPS-limiter increases with limiter temperature, but without any spontaneous peaks (Li-blooms, unipolar arcs e.a.) up to 700^{0}C.
4. D_2 removal from Li limiter can be made on simple heating up to 400^{0}C.

This means, that use of a liquid metal as material of tokamak limiter has no serious physical obstacles.

Laser Inertial Fusion Dry-Wall Materials Exposure to X-rays and Ions* **

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We are investigating the response of candidate first-wall materials under consideration for future Inertial Fusion Energy (IFE) power plants. The materials are separately exposed to multi-pulsed intense ions on RHEPP-1, and to x-rays on the Z facility, both located at Sandia National Laboratories. Details of the exposure conditions have been described previously.[1] The RHEPP-1 accelerator produces ion fluences up to 10 J/cm^2 per pulse, and can expose materials up to 2000 ion pulses per sample. The Z machine fluence from tungsten z-pinch discharges (several J/cm^2 of up to 5 keV photons) is used to expose samples to single-shot x-ray pulses.

Since ~ 10^8 pulses may impinge on a reactor wall over its lifetime, almost no erosion of a flat wall surface per pulse can be tolerated (<1 nm). Since the energy delivery is pulsed, thermomechanical effects such as roughening and surface fatigue can be expected. The materials studied here include primarily tungsten in either pure or alloy form, and graphite and/or carbon composites. Materials are either flat monolithic (single material), or "engineered" materials of 3-dimensional character such as foams or velvets. After exposure to either ions or x-rays, effects on the surface topology and near-surface microstructure are analyzed, and materials response compared with predictions from BUCKY and other modeling codes. Most samples are DC heated at up to 600C, to measure their response under expected reactor wall operating temperatures.

Tungsten in all forms is observed to undergo surface roughening at or below its melting point when exposed to ions. Deep-lying cracks are also observed, evidently due to fatigue. The powder metallurgy (PM) form of W shows the worst tendency for roughening and cracking. The surface morphology can evolve into a complex structure, which can take hundreds of pulses to develop. The roughness can appear to saturate (PM W), or appear open-ended (Ti), or can be minimally present (Cu). In the case of carbon materials, the fiber structure in carbon composites is observed to withstand ion exposure well, whereas the matrix material is readily removed at fluences even below the predicted surface sublimation temperature.

Further multi-pulsed ion exposures to various ion fluences of tungsten/tungsten alloys are planned, both in flat form and 'engineered' geometries. Post-exposure analysis will include surface profilometry, SEM imaging, and XTEM. Also, since roughening of materials treated on RHEPP may be due to plastic flow induced by thermal gradients, the BUCKY computer code is being modified to enable the modeling of plastic flow. BUCKY studies of both W and Ti are planned to better understand material roughening.

* Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed-Martin Company, for the United States Department of Energy under Contract DE-AC04-94AL84000.
** Supported by NRL through the HAPL Program by the U.S. Department of Energy, NNSA, DP.

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Oral Session III-3

Nuclear Technology Experiments and Testing

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Engineering and Physics Assessments of Spherical Torus Component Test Facility

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The results of a broadly based study of the engineering and physics characteristics of the Component Test Facility (CTF) [1] using the Spherical Torus or Spherical Tokamak (ST) configuration [2] are presented. The required testing capabilities [3] of the CTF of high fusion neutron fluxes W_L of > 1 MW/m^2, large total testing area of > 10 m^2, and intense testing fluence of > 0.3 MW-yr/m^2 per year are found to set lower bounds on the CTF size (see Figure). Testing of tritium self-sufficiency further pushes the aspect ratio toward 1.4. A typical CTF design is characterized by R = 1.2 m, A = 1.5, elongation = 3, I_p = 10 MA, B_T = 2.5 T, producing a fusion power of 77 MW and W_L of 1 MW/m^2, assuming moderate normalized ST plasma parameters achievable [4] without active feedback control of MHD modes, while using P_{NBI} = 24 MW at E_{NBI} = 120 kV. Assumption of the advanced physics regime with MHD mode stabilization, while using P_{NBI} = 24 MW at E_{NBI} = 330 kV, would enable W_L = 4 MW/m^2 for testing at the level of demonstration power plants. The ST CTF device requires the use of a single-turn normal conducting center leg for the toroidal field coil without the induction solenoid and substantial neutron shielding. A solenoid-free current start-up RF power of 5 – 10 MW, and a ramp-up and sustainment RF and NBI power of 40 MW are estimated, based on latest data. A new systems code that combines the key physics and engineering requirements, limits, and performance of CTF are prepared and utilized as part of this study. The results show a high potential for a family of CTF devices to suit a variety of fusion nuclear testing and R&D missions. *Support by DOE Contract Nos. DE-AC02-76CH03073 & DE-AC05-96OR22464.

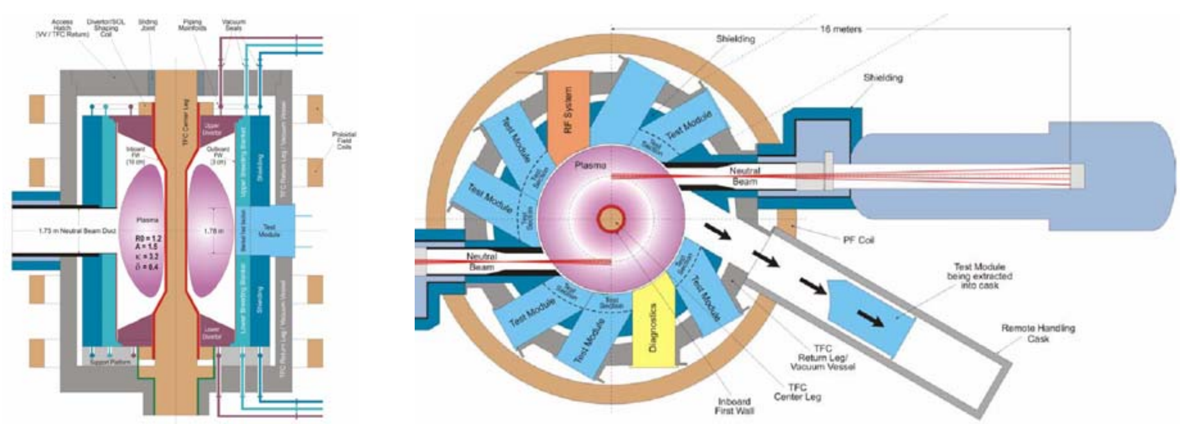

Figure. Elevation and mid-plane views of CTF showing the possible design features dictated by CTF mission.

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An Overview of the Fluid Dynamics Aspects of Liquid Protection Schemes for Fusion Reactors

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This paper provides an overview of experimental and numerical studies conducted at Georgia Tech to assess the fluid dynamics aspects of liquid protection schemes for fusion energy reactors. Problems to be addressed include:

1. Dynamics of slab jets or liquid sheets for thick liquid protection, including the effect of nozzle design, flow conditioning, and boundary layer cutting on jet surface smoothness;
2. Primary turbulent breakup of turbulent liquid sheets and forced thin liquid films, and quantification of the associated hydrodynamic source term;
3. Dynamics of forced thin liquid films on downward-facing flat and curved surfaces, including film detachment and flow around beam ports;
4. Free surface topology and droplet detachment from downward-facing porous wetted walls;
5. Thermocapillary effects and associated design constraints for thin-liquid film protected divertors and first walls; and
6. Inertial fusion energy chamber clearing phenomena, including interactions between liquid drops and an expanding plasma.

Recent Progress of Design & Development in IFMIF Activities

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The International Fusion Materials Irradiation Facility (IFMIF) is an accelerator-based neutron source facility utilizing the deuteron-lithium stripping reaction with an energy spectrum peaked at around 14MeV, to be the best choice for a materials irradiation facility in fusion application with a similarity of the lattice damage production and the nuclear transmutation characteristics. IFMIF is a joint effort currently of the EU, Japan, the Russian Federation and the USA within the framework of Fusion Materials Implementing Agreement of the IEA. A reference conceptual design including a detailed cost estimate and a cost reduction design was developed in the first stage of the activity (1995-99). The primary risk factors of some key component technology to achieve a cw deuteron beam with 40MeV/125mA and to accept a 10MW beam power by a flowing lithium target were reduced by the success of more than 80 tasks performed during 2000-02 as the Key Element Technology Phase (KEP).

The IFMIF conceptual design is based on the use of existing, proven technology to the maximum extent feasible. Thus there are no requirements for basic research to validate any of the principle of technology. However the development activities have been planned to improve or adapt the technologies selected for IFMIF and it was recognized that some development activities and some detailed preliminary design were still required to provide the basis for making a decision on IFMIF construction. In the present schedule a next phase, the Engineering Validation and Engineering Design Activity (EVEDA), is planned to focus on the detailed engineering design and the associated prototypical component tests, under a new organizational structure to allow for enhanced joint team design work and smooth transition to subsequent construction.

In these two years, the IFMIF activity stays in the transition phase from previous phase to the next EVEDA phase, and a working draft for new Implementing Agreement for EVEDA was prepared and a Comprehensive Design Report was published for domestic technical reviews in each party to make a decision to participate the EVEDA. As the preparatory work for starting EVEDA, an extended design study of accelerator beam dynamics, lithium loop experiments, test assembly thermo-mechanical design, etc. have been carried out using an existing resource and a test stand developed in KEP. A new result of end-to-end beam dynamics simulation study shows that a perturbation on output beam energy exaggerates the non-uniformity of beam distribution at the target surface. A lithium flow speed record of 15m/s was achieved at the Osaka Univ. experimental loop and a cavitation signal was observed successfully, however, with no indication of break of flow stability at the nozzle area. Many other technical improvements and design issues clarified through the experimental results are also presented in the talk.
(450 words, limit 500)

Progress on Liquid Metal MHD Free Surface Flow Modeling and Experiments

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The use of flowing liquid metal with a free surface as a plasma contact surface have been the subject of a considerable amount work over the past several years in both the plasma physics and fusion engineering sciences programs. In particular, the ability of some liquid metals, like lithium, to pump hydrogen as well as getter oxygen and water impurities have made the use of *liquid surface technology* of interest to current plasma physics devices as potential particle control tools. In addition, fast flowing free surface liquid metals streams can also remove heat – making their application as a divertor particularly interesting for both near term and future experimental and burning plasma devices. Even for ITER, there is interest in the behavior of liquid metal melt layers in contact with plasma, which will exhibit some similar behavior to intentionally introduced liquid wall or divertor flows.

A key issue for the feasibility of liquid metal plasma contact surfaces is their magnetohydrodynamic (MHD) behavior. Liquid metals are, of course, good conductors of electricity. The electric currents that are self-generated from the liquid motion in the magnetic field, or externally-generated from their serving as closure paths for plasma thermoelectric and halo currents at points of plasma contact, can strongly influence the dynamics and surface shape of the liquid metal flow. Various phenomena are possible depending on the magnetic and flow conditions, including dramatic deceleration of the flow, deformation and deflection of liquid jets, and generation of high-speed droplets torn from the main flow.

This paper describes efforts toward the numerical and experimental simulation of liquid metal free surface MHD flows typical of the use of such flows for liquid divertors, especially in current or near term plasma devices. The 3D incompressible MHD free surface code HIMAG has been applied to several problems of interest. This unique code was developed to allow multiple solid and liquid phase materials with arbitrary geometry to be modeled. The inclusion of complex-geometry, electrically-conducting walls and nozzles are essential since electric current closure paths are typically through these solid structures. HIMAG has been used to analyze MHD flow experiments in the MTOR facility at UCLA and the LIMITS facility at SNL. HIMAG has also been used to analyze liquid metal pools in contact with plasma typical of the DiMES lithium exposure experiments in the DIII-D tokamak. These results are presented and compared with available experimental data. In addition, new experimental studies of wide channel liquid gallium alloy MHD flows in the MTOR facility have been conducted to investigate the effect of highly elongated flow geometry on the flow drag and surface shape. Initial results of this experimental campaign are also presented. Finally, plans for future experimental and numerical simulations in support of the flowing liquid divertor module for NSTX are reported and discussed.

Protection of IFE First Wall Surfaces from Impulsive Loading by Multiple Liquid Layers

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Several inertial fusion energy reactor designs incorporate liquid jets or sheets to protect the first wall and to absorb energy for heat removal from the ICF capsule implosions. These flowing liquid sheets will be accelerated by the hydrodynamic shock and the vaporization due to impact and absorption of the energetic particles. As a consequence, the impulsive pressure load on the reactor first wall is not exclusively due to the hydrodynamic effects of the shock wave from the fusion reaction, but it is also due to the impact of the shock-accelerated liquid layer. This impulsive force along with the hydrodynamics of the break of the liquid protection sheets is studied with the aid of a large vertical shock tube. The shock tube is used to simulate the hydrodynamic shock resulting from the fusion reaction, which makes contact with a water layer suspended in the shock tube. The water sheet is accelerated upon shock contact and begins to break up due to instability growth and vaporization. The degree of breakup and the impulsive end wall force are discussed in this paper. Two different depths of water layer, 6.4 mm and 12.8 mm, are tested at two Mach numbers, 2.12 and 3.20. The pressure histories at various positions along the length of the shock tube and on the end wall are recorded. It is found that the speed of the transmitted shock wave is reduced by about 30% after passing through the liquid layer; however, the peak pressure at the end-wall of the shock tube is significantly increased up to 8 times higher due to the high impulsive force of the liquid layer. X-ray radiography techniques are used to image the breakup of the water layer alloying a quantitative measure of the mass fraction distribution of water after shock impact. It is found that the water layer is significantly disrupted and partially vaporized by the impact even at these low mach number shocks. An apparent mixing and spreading of the water layer is on the order of 20 for a single 12.8 mm initially deep of water layer (4.5 ms after contact with Mach 2.12 shock wave). As many reactor designs involve multiple liquid layers, investigations with multiple water layers are conducted with the goal of understanding the absorption of the hydrodynamic energy in the different layers. It is found that the end-wall peak pressure is indeed reduced up to factors of 2, compared to the single layer configuration with the same amount of water, as multiple sheets with different layer separations are studied.

Z-Pinch Power Plant Shock Mitigation Experiments and Analysis*

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The Z-Pinch Power Plant (ZP-3) is a technological spin-off of the Z Pinch Facility located at Sandia National Laboratories. ZP-3 is a unique, inertially confined, fusion energy concept in which high yield targets will be ignited to fusion, yielding bursts of energy in the 1 to 20 gigajoule range. In particular, a 60 to 100 million-amp pulse in a dynamic Hohlraum results in a magnetic pinch and an extreme X-ray pulse that compresses the deuterium-tritium fuel pellet, resulting in fusion. The fusion reaction yields an energetic burst that consists principally of neutrons, gamma rays, and charged particles. These particles travel through an evacuated space inside the chamber to a series of flibe walls that are separated by additional vacuous regions (~ 2,632 Pa). The particles deposit their energy within the first few mm of the flibe walls. Flibe 1) provides tritium generation for subsequent fusion (via a Li-slow neutron reaction), 2) is a shock wave attenuator, and 3) absorbs neutrons, thus shielding the ZP-3 chamber, which will be designed to survive for 30 to 40 years of operation. In shielding the chamber, the flibe is heated by the highly energetic neutrons. This heat is ultimately extracted by the ZP-3 thermal cycle in order to generate electricity.

In this paper, we will discuss the small-scale shock attenuation experiments being conducted for by Sandia. Our focus will be on the optimization of thick liquid wall streams, with the goal of maximizing shock attenuation as a function of material properties (we may ultimately use a different coolant than flibe), wall thickness, number of walls, and material voiding. In addition, we will also assess the capacity of foamed flibe or some similar material that can fill the entire chamber (as opposed to walls separated by vacuous gaps). We will proceed by benchmarking codes such as BUCKY and ALEGRA with experimental data. We will explore the code's strengths and weaknesses for modeling of the appropriate phenomena. Depending on our findings, new models may be recommended, and if time permits, developed. Finally, we will explore small-scale experiment scalability to the full-scale ZP-3.

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Plenary Session III

The ITER Project

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ITER Status

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At this time the ITER Project still has two candidate sites. One in Rokkasho (Japan, Aomori) and the other in Cadarache (EU, France). A decision is expected imminently and will be taken at the highest political level.
Design work, with much reduced manpower, is underway to develop detailed technical specifications for the long lead items, such as the tokamak building complex, as well as to better define machine interfaces. Engineering and document management tools are also being implemented to prepare for the construction phase and to make use of modern technology.
Once the site will be chosen, the construction of ITER will be a major challenge. In fact, with ITER, the worldwide fusion community will take a large step forward not only from the fusion performance standpoint but also in the engineering and manufacturing quality management necessary to confront the high reliability requirements: the need for remote access and maintenance, component operation in a harsh environment, requirements of licensing, and the sheer size and complexity of the plant, will all require a strong management system that will ensure not only that ITER will be built but that it will also work as planned.
This paper highlights the ongoing design and organizational work as well as some of the challenges that still lie ahead during the ITER construction phase.

US ITER Project Activities

Ned Sauthoff [1]

[1]*US ITER Project Manager*

Since joining the ITER Transitional Arrangements in early 2003, the US has re-engaged in the technical and organizational-planning activities in preparation for an efficient start of construction. The primary focus for the US technical efforts has been completing the R&D and design related to the in-kind contributions that have been tentatively allocated to the US; the US is conducting this work through a combination of assignments to the International Team and conduct of domestic tasks that are synergistic with the scope of the Virtual Laboratory for Technology. The US is emphasizing the management of risk in its preparations and plans. The organizational preparations address the development of US views on the international ITER Organization project management, the establishment of the US Domestic Agency, and planning for effective conduct of the US acquisitions of its in-kind contributions within the context of the USDOE Project Management Order.

*The US ITER Project work is support under contracts with the US Department of Energy.

Relation of US VLT Program to ITER

Dr. Charles C. Baker
Director, Virtual Technology Laboratory

Charles C. Baker received his Ph.D. in nuclear engineering in 1972 from the University of Wisconsin-Madison and has been involved in fusion energy research since that time. He was a fusion energy department manager at General Atomics (1972-1977), director of Argonne National Laboratory's fusion program (1977-1989), program director for fusion technology at Oak Ridge National Laboratory (1989-1994), and head of US work on the International Thermonuclear Experimental Reactor (1992-1999). From 1994 to 2004, he was at the University of California, San Diego, where he continued as head of the US ITER work and served as Deputy Director of UCSD's Center for Energy Research. From 1998 to the present, he has acted as the Director of DOE's Virtual Laboratory for (fusion) Technology and as Adjunct Professor in UCSD's Mechanical & Aerospace Department. He continues in these latter two functions while providing consulting services to Sandia National Laboratories in Albuquerque, New Mexico.

Dr. Baker's professional memberships and activities include: Fellow, American Nuclear Society; Vice Chair, DOE Fusion Energy Sciences Advisory Committee (2002-2004); Principal Editor, Journal of Fusion Engineering & Design (2001-2005); Symposium General Chairperson, 6th International Symposium on Fusion Nuclear Technology (2002).

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