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Overview of fusion nuclear technology in the US

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Abstract

Fusion nuclear technology (FNT) research in the United States encompasses many activities and requires expertise and capabilities in many different disciplines. The US Enabling Technology program is divided into several task areas, with aspects of magnet fusion energy (MFE) fusion nuclear technology being addressed mainly in the Plasma Chamber, Neutronics, Safety, Materials, Tritium and Plasma Facing Component Programs. These various programs work together to address key FNT topics, including support for the ITER basic machine and the ITER Test Blanket Module, support for domestic plasma experiments, and development of DEMO relevant material and technological systems for blankets, shields, and plasma facing components. In addition, two inertial fusion energy (IFE) research programs conducting FNT-related research for IFE are also described. While it is difficult to describe all these activities in adequate detail, this paper gives an overview of critical FNT activities.

Keywords: Blanket; First wall; Tritium; Plasma facing components; Safety; ITER

1. Introduction

Fusion nuclear technology (FNT) as considered in this paper is comprised of all the materials, components, systems and technologies of the plasma chamber that are required to contain, shield, extract energy from, and breed tritium fuel for the thermonuclear fusion plasma. FNT advances will be needed both for near-term magnetic (MFE) and inertial (IFE) fusion energy experiments, and ultimately for MFE and IFE energy-producing power reactors. An incomplete list of FNT components and systems includes:

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- first wall/limiters/divertor, and blanket (breeding and non-breeding);
- conducting shells, ports, antennae, diagnostics embedded in or penetrating through the blanket;
- vacuum vessel, and radiation shielding, support structures;
- structural materials, breeding materials, electric/thermal insulators, tritium barriers, diagnostic windows, armor materials;
- tritium fuel cycle and processing;
- design and integration for chamber components and remote maintenance.

In the United States for the past several years, the base program FNT research and development has been carried out as part of the Enabling Technologies Program supported by the Office of Fusion Energy Sciences (OFES) in the US Department of Energy. The Virtual Laboratory for Technology (VLT) is a multi-institutional, multi-disciplinary research group that oversees and helps guide the R&D activities in the Enabling Technology Program. In addition, the recently created US ITER Project Office (US-IPO) also supports some FNT research activities that relate directly to the US in-kind contributions to the ITER project. The US-IPO exists under the DOE Office of Fusion Energy Sciences and its research is coordinated with the base program R&D. Outside of the DOE Office of Fusion Energy Sciences, however, are two inertial fusion research programs, supported by the DOE Defense Programs Office, that conduct FNT research for inertial fusion. These programs are the High Average Power Laser (HAPL) Program, which has been going on for several years now, and the Z-pinch power plant program, which is in its second year.

All of these programs contribute to the FNT R&D currently underway in the US. Each program, with examples of current R&D, is described in more detail in the following sections.

2. Enabling Technologies Program

The Enabling Technologies Program is coordinated and represented through the Virtual Laboratory for Technology [1]. The mission of the program is to contribute to the US national science and technology base by: (1) developing the enabling technology for

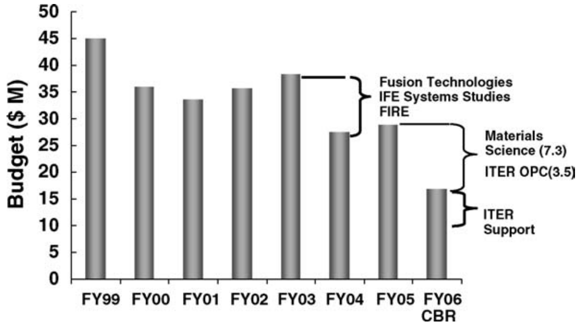

Fig. 1. Recent trends in the US Enabling Technology Program budget.

existing and next-step experimental devices; (2) exploring and understanding key materials and technology feasibility issues for attractive fusion power sources; (3) conducting advanced design studies that integrate the wealth of our understanding to guide R&D priorities and by developing design solutions for next-step and future devices. The Enabling Technologies Program is currently made up of the following subareas for R&D: Plasma Facing Components (PFCs), Plasma Chamber (including MFE and IFE), Materials, Safety & Tritium Research, Tritium Processing, Neutronics, ARIES Design Studies, IFE System Studies, Magnets, Ion Cyclotron Heating, Electron Cyclotron Heating, Fueling, and Socio-Economic Studies. These subject areas and the current R&D being pursued in them are loosely grouped into the categories of Plasma Technologies, ITER Support, Advanced Design, and Materials Research. Only some of the activities can be accurately classified as Fusion Nuclear Technology; various examples will be discussed in the subsections that follow.

The recent trend over the past several years in the Enabling Technologies program has been to reduce long-term reactor relevant R&D, as indicated in Fig. 1. In particular, IFE chamber and design activities and the APEX program [2] for innovative high power density MFE chamber concepts have been concluded, and this year the OFES has requested the closeout of the Materials program in its budget request to the US Congress (note: final US FY06 congressional budget reversed decision to close out materials program this year). Instead, the focus is being placed on technologies and R&D required for near-term plasma experiments, in particular ITER. The following examples give an idea of the current FNT emphasis of the Enabling Technologies Program.

2.1. ITER Shielding Blanket Module 18 and ITER PMI

The US has a longstanding interest in developing PFCs, including plasma facing and heat sink materials and fabrication technologies, testing the power handling capability and fatigue lifetime of divertor and first wall components, and interaction of the edge plasma with material surfaces. As part of its contribution to ITER, the US will develop the design of ITER FW Module 18—the lowest outboard module just above the divertor. Mod18 is unique from other FW modules in that it is mounted on the triangular support, an appendage on the vacuum vessel wall, is thinner (400 mm versus 450 mm) than other modules, has various port penetrations, and part of its lower surface in addition to the front face is exposed to the plasma (see Fig. 2). The FW's CuCrZr heat sink must be joined to beryllium armor, internal cooling channel liners, and a return manifold of 316LN-IG [3]. A key issue is eddy current control and determination of the number and position of cuts in the metal block. Model development and analysis is underway with the OPERA® code interfaced with solid modeling in CATIA and temporal and spatially dependent plasma current data output from the Russian DINA code used for plasma disruption simulation.

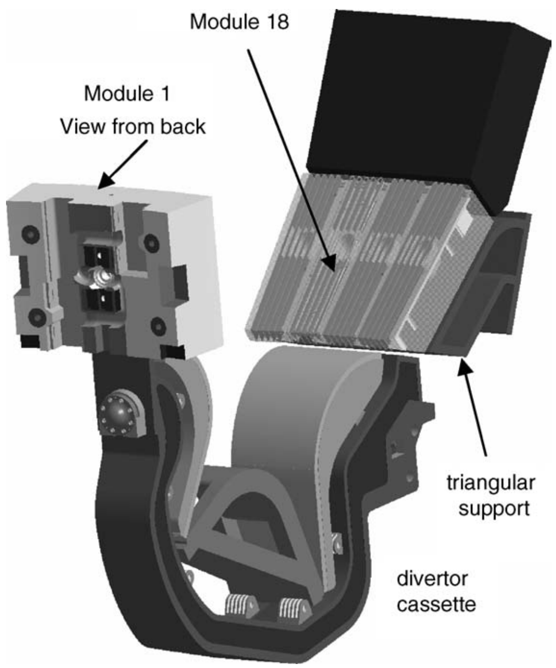

Fig. 2. CAD model of Module 18 in position above the ITER divertor segment; view is looking toward outboard face from inside.

The US is also involved with studying mechanisms of erosion, transport and deposition in steady and transient plasma conditions that are still not well understood, especially for mixed materials. Recent work on Be seeding in incident plasma shows that the Be tends to cover up the graphite surface, thereby reducing the erosion of carbon and the danger of carbon and tritium codeposition [4]. Other work on experimental and numerical simulation of ELM loads and convective "blob" cross-field transport is also underway.

2.2. Advanced liquid plasma facing surfaces

Over the past 6 years, two innovative blanket and PFC technology programs, APEX and ALPS [5,6], investigated the feasibility of several high power density reactor ideas utilizing liquid walls and free surface divertors. This effort is now highly focused on developing a liquid lithium free surface divertor experiment for fielding in the National Spherical Torus eXperiment (NSTX) device. The purpose of such an experiment is to provide a particle and heat load control tool for long pulse NSTX discharges. The effort is divided into a staged approach, where the first stage involves introducing a system to coat PFC tiles in NSTX with a ∼300 nm layer of lithium between each discharge in order to observe the effectiveness of the lithium surface on hydrogen and impurity particle control. Ultimately, the goal is develop a fast flowing lithium system with a significant fraction of square meter surface area to rapidly pump and remove particles and energy.

Plans to deploy the various free surface modules have been slowed this year due to increasing effort on ITER R&D needs; however, tasks for the advanced liquid plasma facing surfaces are continuing. Critical issues include developing lithium coating systems and determining the surface conditions of lithium coatings on carbon and other PFC surfaces. Magnetohydrodynamic effects are very significant for the fast flowing system and both strong simulation and experimental investigation of prototypic lithium free jet and film

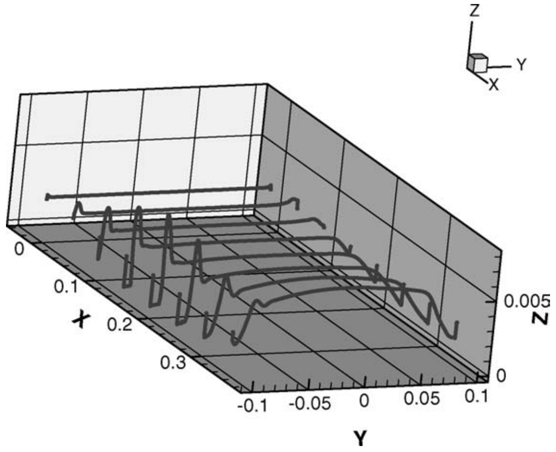

Fig. 3. Surface contour of lithium free surface flow in NSTX-like magnetic field showing asymmetric pinching effect. Initial depth 2 mm with a velocity 10 m/s at an uphill incline of 22°.

flows is a key task [7]. For example, Fig. 3 shows the distortion and deceleration of a liquid lithium film flow in a typical NSTX-like magnetic field.

2.3. ITER Test Blanket Module (TBM) experiments

Utilization of the fusion environment for performing increasingly integrated first wall and breeding blanket experiments has always been an important ITER mission element. Since the US decision to rejoin ITER, the Plasma Chamber community, together with significant participation from the PFC, Safety, Materials, and Tritium programs in the US, has participated in the ITER Test Blanket Working Group (TBWG) and has proposed to develop, in collaboration with other interested ITER parties, solid lithium ceramic and liquid lead–lithium breeder blanket experiments for ITER [8]. The US strategy is to emphasize strong international collaboration on these two classes of first wall/blankets with the other ITER partners and members of the International Team to develop the concepts, designs, test plan, responsibility sharing, safety portfolio, and ITER operation and machine interface needs necessary for successful testing in ITER. Work is underway to prepare the Design Description Document for each concept, including significant design analysis in areas such as mechanical and hydraulic behavior, EM analysis, neutronics, safety, diagnostics, tritium systems, etc.

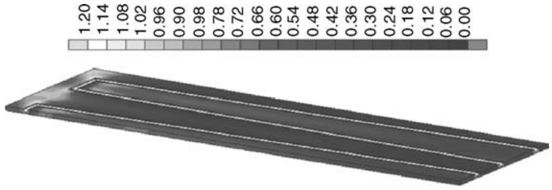

Fig. 4. Calculated Von Mises stress profile in a TM unit cell (maximum stress of 1.2 MPa occurs inside the Be bed).

2.3.1. Solid breeder TBMs

All ITER parties are interested in studying aspects of helium-cooled first wall and ceramic breeder blankets. The US plan is to develop specific breeder units, placed inside the EU helium-cooled pebble bed (HCPB) Test Blanket Module that address particular areas of US interest and expertise—the thermo-mechanical performance of lithium-based ceramic and beryllium pebble materials over periods of time that extend to the full lifetime of the component. The main foci of the pebble bed thermomechanics work currently underway involve the development of experimentally validated predictive capabilities to address:

- time-dependent thermomechanics interaction and corresponding stress and strain deformation histories of ceramic breeder pebble beds, including finite element simulations and micromechanical models, as well as empirical consecutive correlations (see, for example, Fig. 4);
- cyclic effects on the integrity of the pebbles and dimensional stability of the beds at the interface, and the modeling of the interrelationship between the formation of the interfacial gap and subsequent temperature and stress responses.

2.3.2. Dual-coolant lead–lithium TBMs

The US technology community is also proposing to further develop a dual-coolant He/lead–lithium (DCLL) [9] first wall and blanket concept (originally developed as part of the ARIES-ST study [10]). The US strategy here is to attempt to improve the performance of the similar EU helium-cooled lead–lithium (HCLL) concept by the addition of a high temperature PbLi self-cooled breeding zone electrically and thermally insulated from the ferritic steel structure by SiC/SiC composite flow channel inserts [11]. Once again, strong collaboration is desirable with the EU community, who are already addressing many of the

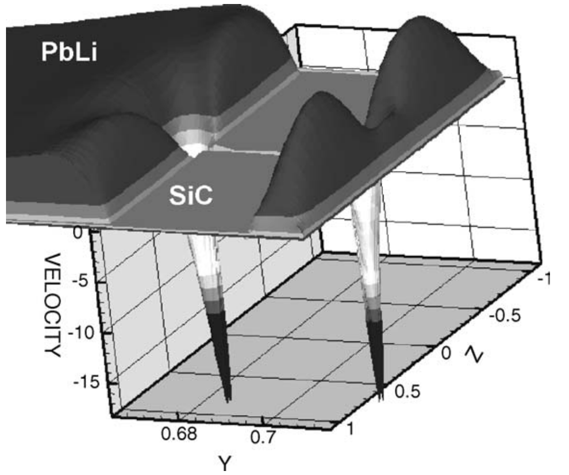

Fig. 5. Reversed MHD flow jets near crack in SiC flow channel insert (velocity normalized to channel average).

R&D activities required for both the HCLL [12] and DCLL concepts, such as ferritic steel fabrication, tritium extraction, PbLi corrosion, etc. and with new ITER partners (for instance China) who also have interests in pursuing similar PbLi concepts. The unique area of US interest is the magnetohydrodynamics interaction and compatibility of the flowing self-cooled PbLi zone with the SiC/SiC composite flow channel insert. Near term R&D is focused on:

- MHD and heat transfer analysis flows in poloidal channels and manifold regions, and the function of FCI as thermal and electrical insulator (see Fig. 5 [13]).
- SiC composite electrical and thermal properties, fabrication techniques, and experiments with PbLi in static compatibility tests.

The need for flow loop testing of MHD effects and PbLi compatibility with FCIs is envisioned in the near future.

2.3.3. Tritium removal from TBM coolant streams

Both the ceramic breeder and PbLi breeder have issues associated with permeation of tritium from the breeder region or through the first wall into the helium coolant streams. It is not at all desirable to cool these streams to cryogenic temperatures in order to remove tritium via molecular sieves, and instead a palladium/silver alloy permeator operating at 300–500 °C is considered for this separation. By maintaining a vacuum on the permeate side of the permeator, it was shown that even small amounts of hydrogen isotopes could be effectively removed from a stream of helium [14], and this was found to be attractive relative to other separation techniques [15] Such a permeator may also be usable for removal of tritium from hot PbLi as well if more conventional direct contact stripping in bubble columns proves to be too inefficient.

2.3.4. Flibe chemistry control and MHD effects on turbulent heat transfer

During the early phases of the US ITER TBM, some consideration was given to the possibility of testing a BeF_2–LiF-based molten salt (referred to here generically as “flibe”) in a self-cooled or dual-coolant blanket configuration. Flibe was being considered in the APEX study both for liquid walls and closed channel blanket designs [5], and it was also selected as the working medium for thick liquid protection jet arrays for Heavy Ion Fusion IFE reactor concepts [16] and more recently for the Z-pinch IFE reactor (discussed below). While the PbLi concept was selected over flibe as the main focus of the US effort, the flibe idea was retained as a backup option, and critical R&D on flibe chemistry control using REDOX agents and MHD effects on flibe turbulent heat transfer, both being performed under the JUPITER-II US DOE/Japan MEXT collaboration, is continuing through the end of the current collaboration agreement in April 2007. Results on REDOX control have shown that Beryllium metal may be a good REDOX control agent due to its relatively significant solubility in flibe and its effectiveness in reducing free HF and F following nuclear transmutation of flibe (see Fig. 6). MHD heat transfer and turbulence visualization experiments utilizing a aqueous KOH solution as a flibe surrogate are just now getting underway after good benchmarking of turbulence and heat transfer data acquisition systems against past results and state-of-the-art turbulence codes (see Fig. 7) [17].

2.4. Neutronics CAD/MCNP coupling

Currently, domain models are input to 3D neutronics codes such as MCNP [18] by specifying geometric surface definitions and combinations of those surfaces to give 3D solids. The 3D geometric models typically used with MCNP are only rough approx-

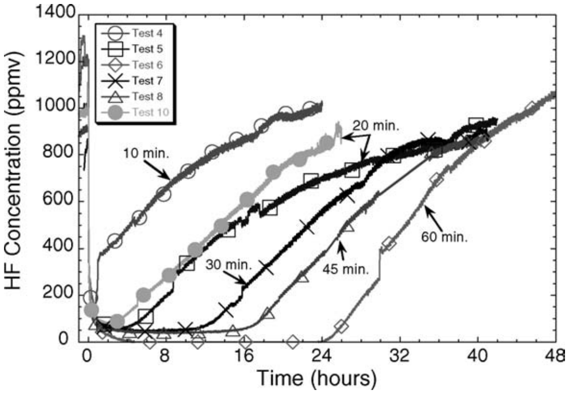

Fig. 6. Transients in HF concentrations measured after Be insertion/removal from flibe at 803 K. HF input concentration was 1000 ppmv for all tests and significant time to return to this concentration indicates HF being reduced by dissolved Be.

imations of the actual as-built geometry, with fine geometric details missing from the model. Building complex models for MCNP can be quite cumbersome, and each model must be re-constructed manually to account for any changes. For complex and rapidly changing designs anticipated for ITER (or other burning plasma experiment) and the Compact Stellarator [19] in the US, this can become time consuming and error-prone. Currently under development is the capability to evaluate the exact CAD geometry by linking the CAD modeling engine directly into MCNP. The approach uses the Common Geometry Module (CGM) [20] to interface directly with CAD geometry engines.

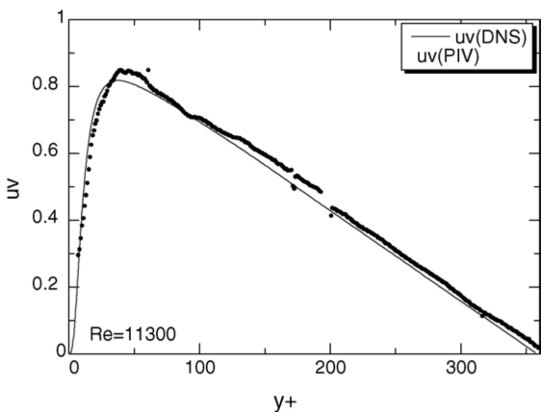

Fig. 7. Agreement between measured and calculated Reynolds stress values in turbulent flibe-simulant pipe flow as a function of distance from the wall.

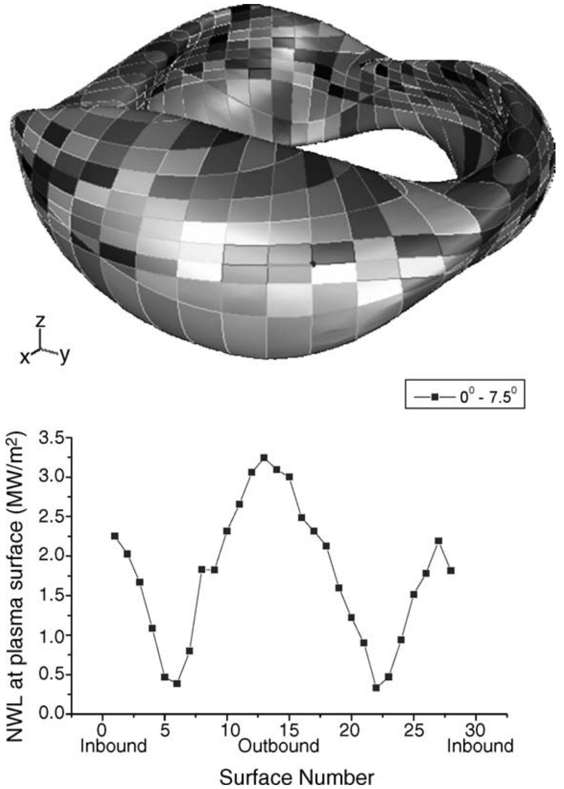

Fig. 8. Computation model for ARIES-CS and calculated wall loading distribution.

A proof-of-concept integration of CGM into MCNP has been implemented and verified against the unmodified MCNP code. Modifications have been made primarily to the ray-tracing functionality in MCNP, which finds the next intersection of a particle with a geometric surface and the cell on the other side of that surface. This code is being used to study first wall loading in the compact stellarator design ARIES-CS as shown in Fig. 8 [21]. The immediate focus of this research is to accelerate the ray-tracing portion of the code.

2.5. Irradiation effects in materials

The longer-term materials development effort in the US is focused on low-activation materials for structural applications. Ferritic/martensitic and oxide dispersion strengthened (ODS) steels are the most mature on a scientific and engineering basis, and are the leading candidate for first-generation demonstration fusion power systems. Vanadium alloys offer the potential for high thermodynamic efficiency, but have an incomplete database and require strict atmospheric control during processing, joining, and operation. SiC composites offer the highest potential operating temperature, but have the greatest development risk and least technological maturity. However, the nearer term potential of SiC/SiC composites as flow channel inserts, which require neither high strength or high thermal conductivity, but which do require some radiation resistance and a good database for irradiated thermophysical properties, help motivate continued development and radiation effects testing.

Recent research has demonstrated [22,23] the feasibility of producing ODS steels containing Y–Ti–O nanocluster densities on the order of $10^{24}\,m^{-3}$ with cluster diameters of about 5 nm. Annealing studies [22,23] have shown that these nanoclusters possess high thermal stability at anticipated operating temperatures of ∼800 °C. Computational studies are underway to develop a multi-scale model of He transport and fate in ODS steels. Molecular dynamics (MD) simulations are being employed to explore the trapping and migration of He in microstructural features such as dislocations, grain boundaries and coherent nanoclusters. Such studies have revealed that He is strongly bound to grain boundaries and the binding energy correlates with grain boundary excess volume [24]. Fig. 9 gives recent results of interstitial He diffusivity in the matrix and in grain boundaries derived from elevated temperature MD simulations [25]. Note that interstitial He diffuses much more slowly in grain boundaries than in the matrix at projected operating temperatures.

The importance of this work is internationally recognized and is the subject of the ongoing DOE-JAERI collaboration on irradiation effects in ferritic steel materials. Other irradiation experiments are currently underway or upcoming on a variety of vanadium and SiC composite samples, also the subject of international collaboration under the Jupiter-II DOE-MEXT collaboration.

3. US ITER Project

This year the PPPL-ORNL proposal to run the US ITER Project Office (US-IPO), the legal entity that will lead the ITER effort in the US, was selected over two competing proposals. The DOE FY06 congressional budget request has asked for significant funds to allow the US-IPO to prepare for procurement of US in-kind contributions, including R&D, design, fabrication, and oversight; and to support US staff for the ITER organization (secondees). The details of the US-IPO activities are summarized by Sauthoff [26]. FNT R&D activities supported from the US-IPO include most notably:

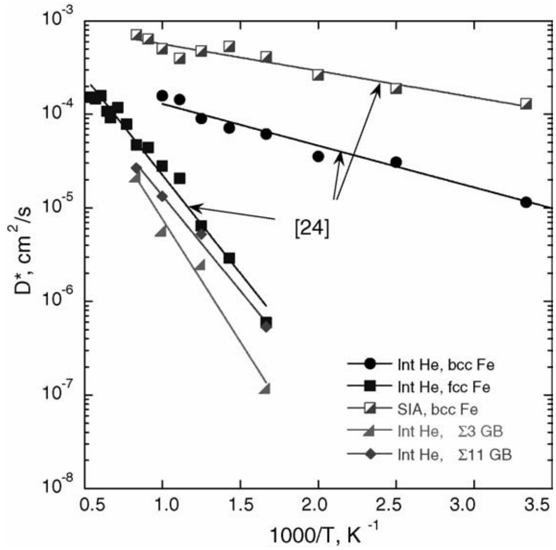

Fig. 9. Inverse temperature dependence of interstitial He diffusivity in Fe matrix and grain boundaries compared to the diffusivity of self-interstitial atoms (SIA).

- FW/Shield Module 18 (described above).
- Participation in diagnostic port development and port plug engineering.
- Developing details of the tritium plant.
- Screw extruder pellet injector.
- Tokamak dust characterization, mobilization and transport calculations.
- Design integration including aspects of CAD, neutronics, thermal hydraulics, EM analysis and materials.

There is still overlap in these tasks with the base program Enabling Technologies R&D activities while the coordination of these efforts is just emerging. It should also be noted that areas like the Test Blanket Module Program are not officially considered under the procurement packages but will require some sort of international agreement and representation in the US-IPO to coordinate this work with ITER.

3.1. One example: Tokamak Exhaust Plant

The US has provisionally been assigned responsibility for the Tokamak Exhaust Processing (TEP) system—one of the Tritium Plant procurement packages. The main purpose of this system is to recover hydrogen isotopes, including those bound in molecules such as water and methane, and deliver purified hydrogen isotopes to the isotope separation system. The ITER TEP technologies include permeators, reforming reactors and isotopic exchange reactors [27].

The TEP construction activity will begin with finalizing the detailed design, and it is expected that industry will fabricate the system. The TEP must process a flow rate approximately 10 times larger than any previous system, and it will have an inventory approximately 10 times larger than previous systems. Largely because of this, it is expected that the TEP will undergo a multi-year testing program prior to being put into production. Dynamic computer modeling is being considered to predict the behavior of this system and mitigate the risk associated with its rather large scale-up. The TEP is highly integrated with other Tritium Plant subsystems, so close interactions will be needed with other procurement package owners and with the ITER International Team.

4. High Average Power Laser and Z-pinch Programs

The High Average Power Laser (HAPL) program, supported by the DOE Defense Projects Program is carrying out a coordinated and focused effort to develop Laser Inertial Fusion Energy (Laser IFE) based on lasers, direct drive targets and a dry wall chamber [28]. While a large portion of this total effort is directed to the development of the laser drivers (Krypton Fluoride Gas Laser and Diode Pumped Solid State Laser), final optics and to solving issues linked with the design, fabrication and injection of the direct-drive target, there is also an appreciable effort on chamber and materials (in particular the first wall and armor) that is certainly related to Fusion Nuclear Technology.

IFE operation is cyclic in nature and the wall is subjected several times per second to prompt energy deposition from the X-rays and ions produced by the fusion micro-explosion. Most of this energy deposition occurs close to the surface, and only a thin layer of the first wall is subject to high temperature cycles, while the rest of the first wall and the blanket at the back essentially see steady state operation. This was the basis for the selection of a configuration with a thin armor providing the threat-accommodation function over a first wall providing the structural function, with tungsten and ferritic steel as preferred armor and structural material, respectively. These points are illustrated in Fig. 10, which shows the temperature response of such an armor configuration for an example 154 MJ yield direct drive target in a chamber of radius 6.5 m [29]. Most of the temperature transients occur well within 100 μm of the surface, while regions beyond 0.1–1 mm are subject to quasi steady-state conditions, akin to the MFE case. This also allows for the possible use of a number of compatible blanket designs that are being developed for MFE; thus, the IFE R&D resources can be more effectively directed to solving the IFE-specific armor/first wall issues, while maximum use can be made of all the information available from the large worldwide MFE effort on blankets, including testing in the fusion nuclear environment of ITER.

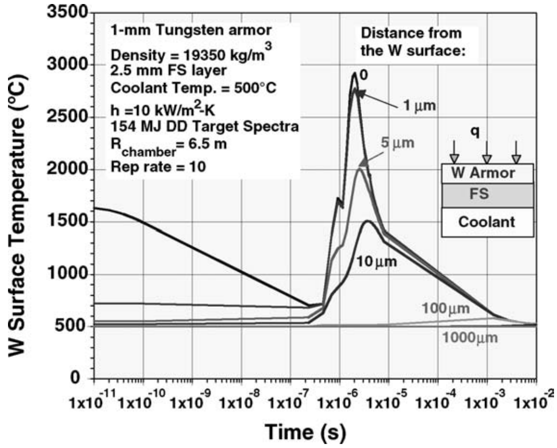

Fig. 10. Temperature history at different locations in a 1 mm W armor over a 2.5 mm FS substrate cooled by a 500 °C coolant based on the power deposition for a 154 MJ direct drive target spectra and assuming a chamber radius of 6.5 m and no protective gas in the chamber.

A key issue is survival of the tungsten armor under the cyclic X-ray and ion threat spectra. Several possible mechanisms could affect the armor survival,

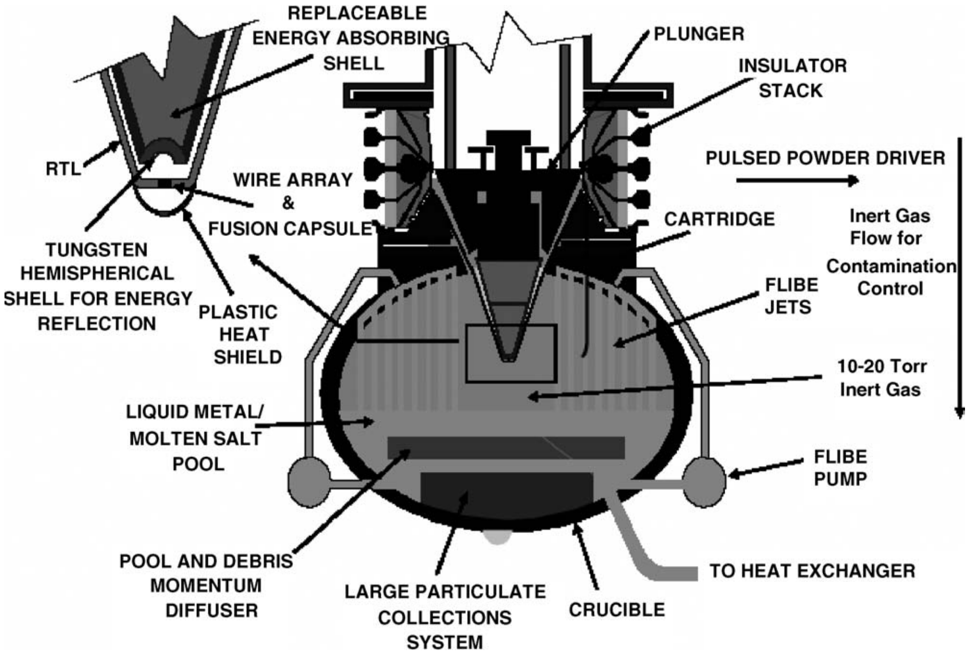

Fig. 11. Chamber concept for Z3 Z-pinch reactor concept.

including: failure of the armor due to ablation, melting, surface roughening and/or fatigue (due to cyclic thermal stresses); failure of the armor due to the accumulation of implanted helium; and failure of the armor/substrate bond due to fatigue. The HAPL design and R&D effort in the chamber and material area is focused toward understanding and resolving these three major issues [30]. The effort includes modeling and experimental testing of the armor thermo-mechanical behavior in facilities utilizing ion, X-rays and laser sources to simulate IFE conditions. Helium management is addressed by conducting implantation experiments along with modeling of He behavior in tungsten. Significant progress has been made recently toward solving the latter two issues (helium retention and bond fatigue), but the first (long term survival of the armor) remains a key unresolved issue. The possibility of utilizing an engineered porous armor is also considered to help in accommodating thermal stresses and in enhancing the transport of implanted helium back to the chamber by minimizing the He diffusion length within the solid W.

The Z-pinch Power Plant (ZP3) program is a collaborative project initiated in 2004 by Sandia National Laboratory to investigate the scientific principles of a power generation system based on a Z-pinch driver system for IFE [30,31]. The current concept of the ZP3 power plant, shown in Fig. 11, is to use the Z-pinch accelerator to compress a DT target resulting in the emission of about 3 GJ of energy mostly carried by neutrons and X-rays. The target is coupled to the pulsed power system through a recyclable transmission line (RTL). The energy released is absorbed by a thick curtain of flowing flibe molten salt, which serves as a heat transfer agent and tritium breeding medium. The power plant is operated in a repetitive mode, with approximately one cycle every 10 s per chamber. Several critical fusion nuclear technology issues exist for a Z-pinch-based IFE power plant, including the energy adsorption and shock generation and mitigation in the flibe curtains, and the destruction and reformation of the RTL.

Contrary to previous HIF-IFE chamber designs utilizing thick liquid wall protection, the ZP3 concept does not need complex liquid geometries for driver penetrations as the energy is transmitted to the target by direct contact. This also eases the requirements on the vacuum level inside the chamber, which is designed to operate with 10 Torr of background Argon. However, the inner gap of the RTL must operate in high vacuum, so a sliding seal arrangement has been devised to allow pre-pumped RTLs to be connected to the main power flow without opening the full chamber to high vacuum environment between each shot when the new RTL is inserted. In Z-pinch IFE there are essentially three configurations of protection that need to be addressed: a conical top plug containing flibe shields the top, flowing liquid layers protecting the side walls, and a two-phase (bubbling) pool at the bottom of the chamber. Studies concerning the flowing liquid layers in shock tube experiments verified that several layers separated with a void significantly reduce the wall loading [32], and there may exist an optimum void-space-to-liquid-thickness ratio to minimize the impulse force and reduce the liquid sheet break-up. Work is ongoing to study and model the shock attenuation and energy absorption in solid foam layers and in two-phase pools for the upper and lower protection schemes proposed in Z-pinch reactor designs.

The RTL will be destroyed on each shot, the remnant will be removed, and a new RTL will be inserted for the next shot. This can be done either utilizing an RTL composed of frozen flibe, which would then melt into the coolant, or a material immiscible in the molten salt, which could then be separated and recycled. A large amount of excited flibe vapor is generated between each shot by the absorption of the X-ray flux generated in the target explosion by the thick liquid wall. The prediction of flibe vapor dynamic in the chamber is necessary to design the RTL insertion mechanism as well as to correctly predict the interaction of the steel RTL fragments with the liquid surfaces. Pulsed power facilities have been used to generate excited flibe vapors from a pool of molten salt in a sealed chamber maintained at 600 °C and filled with different background pressures of argon. Experiments are now underway to characterize the interaction of carbon steel vapors generated with the exploding wire technique with a pool of molten flibe. The objective of the experiments is to characterize the distribution of steel droplets in the liquid, as well as the fraction of metal vapor that reacts with the salt vapors on the surface.

5. Conclusions

A wide variety of fusion nuclear technology R&D activities are currently underway in the US, varying from evolutionary progress on improving neutronics codes, increasing understanding of MHD flow phenomena, pebble bed thermomechanics behavior, and transmutant helium behavior in materials to new design and analysis efforts on ITER shielding blankets, ITER Test Blanket Module experiments and IFE reactors.

The emerging importance of the ITER basic machine in the efforts of the US Office of Fusion Energy Sciences Enabling Technology program is readily apparent in the selection of R&D activities described in this paper. Long term reactor relevant R&D efforts have been shifted and focused to those first wall and blanket concepts and materials that will be tested in ITER, with a strategy of maximizing international collaboration and consolidation of effort. IFE FNT R&D programs have been shifted to other funding sources in Defense Department programs.

There are major concerns among the US scientists and engineers that the recent policy trend of eliminating research on “long term” technologies and technical issues will have negative consequences on the ability of the US fusion program to realize its goal of demonstrating the potential of fusion as a viable and attractive energy source for many decades to come. Despite these concerns, the enthusiasm and commitment of fusion nuclear technology researchers in the US remains strong, owing to the prospect of contributing to ITER and utilizing the ITER fusion environment to advance the understanding and development of fusion nuclear technology.

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