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PROGRESS IN FUSION TECHNOLOGY IN THE U.S. MAGNETIC FUSION PROGRAM *

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Much progress has been made in the last two years in the U.S. magnetic fusion technology program. Notable advancements have been made in superconducting magnets, gyrotrons, reduced activation materials, pellet injectors, and safe handling of tritium to name a few.

Advancements in fusion plasma conditions and economic, environmentally attractive reactor designs will continue to require significant improvements in the technology of magnets, heating, fueling, materials, and safety. The ITER process will provide a forum and an opportunity to accelerate the development of fusion technology through new international collaborations, and we should take advantage of that opportunity.

1. Introduction

The challenges in the area of fusion technology development have greatly increased in the last two years. The design requirements for CIT (the U.S.–Compact Ignition Tokamak) and ITER (International Thermonuclear Experimental Reactor) set very significant demands on the U.S. fusion technology program. At the same time, progress in fusion technology in the U.S. Magnetic Fusion Program over the last two years has been significant, and a considerable number of mutually beneficial international cooperative efforts are underway.

Advancements have been made in plasma heating systems and in validation of the data base in superconducting magnets. The very successful completion of the International Energy Agency (IEA) Large Coil Task confirmed the ability to construct and operate superconducting magnets singularly and in a toroidal geometry at fields to 9 Teslas. Two high power Ion Cyclotron Range of Frequencies (ICRF) antennas were fabricated and installed on TFTR, and a similar one has been constructed for installation on Tore Supra. In the millimeter source area, industrial capability to manufacture high power 140 GHz gyrotrons operable in the pulsed mode up to 800 kW, in addition to the previously existing capability of producing 200 kW cw devices, has been established. Pneumatic pellet fueling injectors have also been fabricated and placed in operation on TFTR and JET, and a centrifugal injector capable of providing pellets at a rate of 10/s for tens of seconds is presently being tested before shipment to Tore Supra.

We have completed a study (ESECOM) which assessed the environmental safety, and economic aspects of fusion to help the U.S. program better focus its R&D efforts. In addition, we have made progress in developing low activation materials and improved coatings and protective tiles for plasma-facing components. Pumped limiters have been developed and supplied to TEXTOR and Tore Supra, and other nuclear technology and materials cooperative efforts are underway.

In this presentation, we will report on the specific progress made in fusion technology in the U.S. Magnetic Fusion Program as organized under the program areas of *Plasma technologies, Fusion technologies,* and *Fusion design studies.*

2. Plasma technologies

The Plasma Technologies Program supports the activities in *Magnetic systems, Plasma heating,* and *Plasma fueling and particle control.*

2.1. Magnetic systems

The goal of the magnet systems technology program is to establish a technical base for efficient magnet

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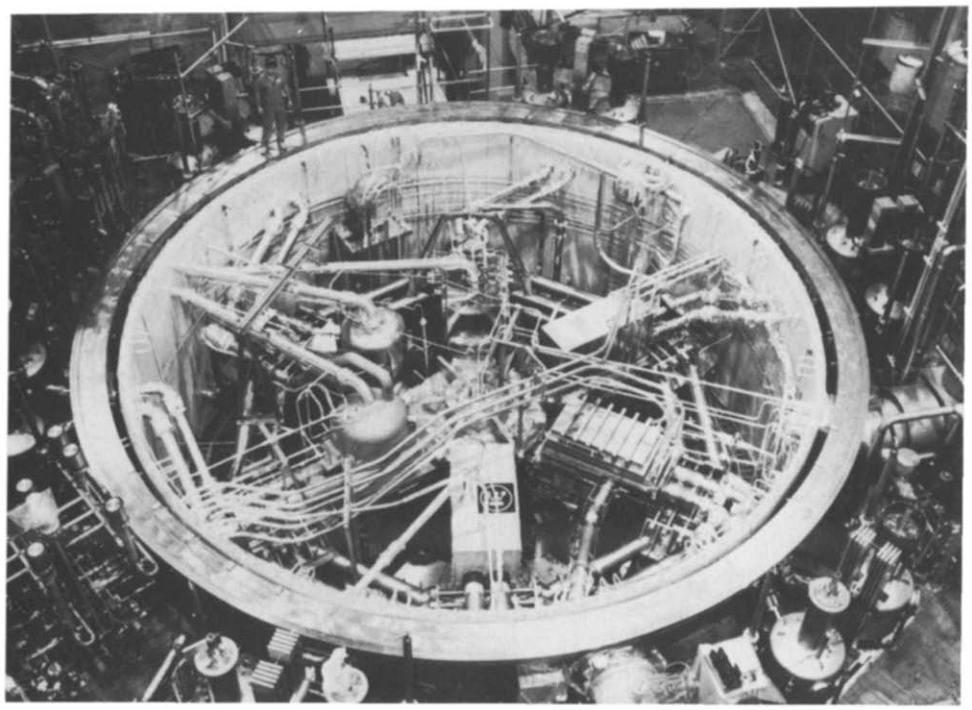

Fig. 1. The Large Coil Test Facility at ORNL showing the six 8 T superconducting magnets.

configurations for plasma shaping and confinement. This necessitates the development and demonstration of scalable and cost-effective high field and pulsed magnet systems. The elements of this program include research and development on superconducting magnet components/subsystems, magnet materials, and cryogenics.

The development and demonstration program has included the Large Coil Task (LCT), high field magnet development (10–15 T) and pulsed magnet development (> 10 T and 1 to 10 T/s). The current focus of the magnet program is on providing the supporting R&D for validating the ITER magnet design assumptions.

A major activity completed in September 1987 was the test program of the IEA Large Coil Task that was conducted at Oak Ridge National Laboratory (ORNL). This program, which was one of the largest international collaborations in the IEA, involved the testing of six 8 T superconducting magnets. In addition to providing the test facility, the U.S. supplied three of the test coils. The other participants, Japan, Euratom, and Switzerland, each provided one test coil. The six coils were tested in a toroidal array in the facility as shown in fig. 1. All of the coils were tested to their design point (8 T) and to their maximum field and current. Additionally, coils were tested with the pulsed coils system, providing informative data simulating tokamak operating conditions. All coils were tested simultaneously as a symmetric torus to a record-setting field of 9 T. Further tests were successfully run to investigate safety aspects of the coil systems. The test results, including data on structural and electromagnetic responses from a variety of conditions, have been analyzed, and a special issue of the journal of "Fusion Engineering and Design" has provided the final report for this project [1].

The high field and pulsed magnet research and development activity is currently being pursued at Lawrence Livermore National Laboratory (LLNL), Massachusetts Institute of Technology (MIT), and ORNL. The tasks involve the design, fabrication, and testing of various superconductors and coils with primary emphasis on the development of Internally Cooled Cabled Superconductors for future use in pulsed high field central solenoids, shaping coils, and toroidal field magnets such as those envisioned in the designs for ITER [2]. High field tests are planned on a subsized pancake module (14 T) in the coming year at LLNL as part of the development process. In addition, a pulsed pancake module is being designed and fabricated by MIT that will be tested in the Superconducting Engineering Test Facility at the Japan Atomic Energy

Research Institute (JAERI). Other related development efforts include the investigation of coil protection, stability, heat generation and removal schemes, development of low loss joints and terminations, and alternate conductor designs.

2.2. Plasma heating

In addition to ohmic heating, energetic neutral particle beams and high frequency electromagnetic waves have been shown to be capable of heating plasmas to thermonuclear temperatures. The many variations of techniques within each method, the complexities associated with the technology, and the physics of plasma interactions make it difficult to choose a reference at this time. Hence, component development and experimentation in both areas are systematically being pursued.

Long pulsed (20–30 s) positive ion based 80 and 120 keV neutral beam systems have been developed for current experiments. Long pulse sources manufactured by RCA were installed on TFTR and D-III in 1987 and have operated very reliably [3]. TFTR has exceeded the design specifications by injecting 28 MW of neutral power into the plasma. Since the efficiency of the positive ion based system falls off at higher particle energies, negative ion neutral beam systems are being developed for use up to 1 MeV and beyond. Worldwide results on the use of neutral beams for current drive have caused a resurgence in the interest of developing negative ion based neutral beams and the selection of them for the ITER design [4].

A conceptually more attractive system than neutral beams for heating plasma is electromagnetic waves with frequencies ranging from megahertz to over a hundred gigahertz. Conventional RF sources in the MHz range of frequencies at the desired megawatt power levels are readily available from industry. These sources are used for heating in the Ion Cyclotron Range of Frequencies (ICRF).

The development efforts in the ICRF range of frequencies have been primarily directed towards reliable wave launching antenna systems which can handle high power and high power density. The development efforts are concentrated on compact antenna systems that can be inserted into the plasma chamber directly through a standard port. Both loop and waveguide variants of the compact antenna are under development [5]. These an-

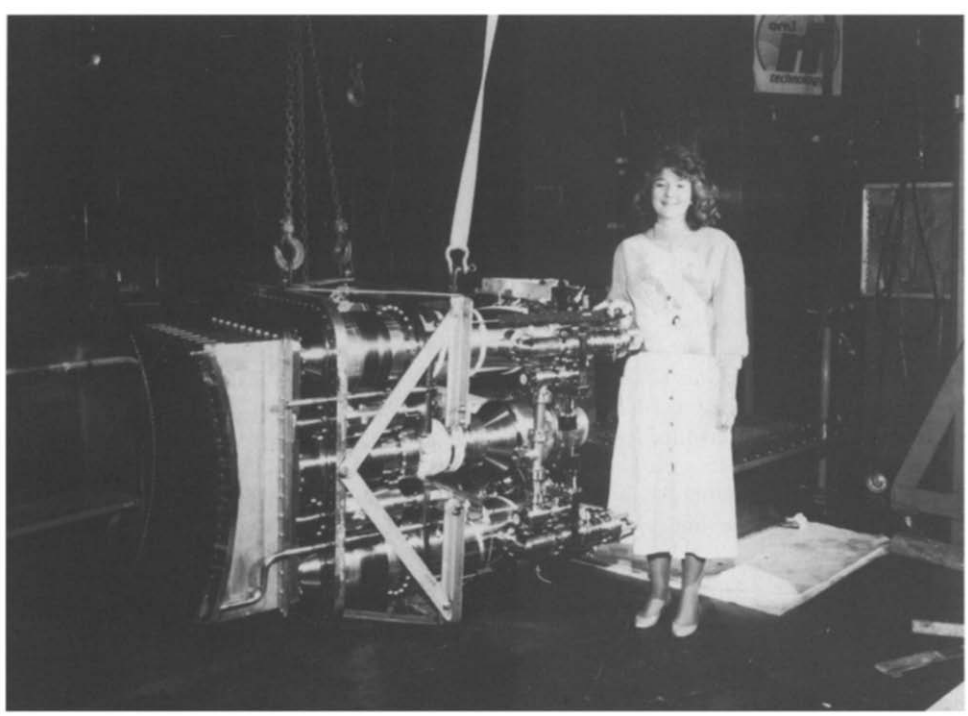

Fig. 2. The 4 MW ICRF antenna developed at ORNL for TFTR an Princeton.

tenna are the same size as the port so that they do not require internal access for installation. They can be moved radially relative to the plasma edge. A further advantage is that they can easily be removed for maintenance. The Radio Frequency Test Facility (RFTF) at ORNL is used to test these development antennae in an actual plasma environment. A compact antenna for TFTR shown in fig. 2 has been developed and tested in RFTF and installed on TFTR [6]. One for Tore Supra is presently under test. One somewhat surprising finding is that some of the Faraday Shield designs in use today are rather inefficient in transmission of the RF energy to the plasma. Hence, we are developing rugged and reliable Faraday Shields using more efficient designs that have been demonstrated. We are also starting to look at the possibility of using the range of ICRF frequencies for fast wave current drive.

Heating plasmas confined by magnetic fields up to 10 T by means of Electron Cyclotron Resonance would require high power millimeter wave sources with frequencies as high as 280 GHz [7]. For more than a decade, the U.S. program has been supporting R&D efforts to establish an industrial base which could supply high power gyrotrons for fusion use. Currently, Varian, the major gyrotron manufacturer in the U.S., can provide steady state and pulsed 200 kW gyrotrons in the 28 to 70 GHz range as well as steady state 100 kW 140 GHz gyrotrons. Industrial development of a steady state 0.5 megawatt 140 GHz gyrotron is in progress. Fig. 3 shows a 400 kW, 140 GHz cw gyrotron built by Varian and satisfactorily tested to 823 kW in a pulsed mode. Steady state testing of the device is now in progress. Efforts are now shifting toward developing one megawatt 140 and 280 GHz, cw gyrotrons. During the past year the MIT program has demonstrated pulsed operation of a whispering gallery 750 kW gyrotron operating at a frequency of 140 GHz and stepwise tunable up to 243 GHz. The first test of a high power quasi-optical gyrotron operating above 100 GHz and tunable over a 10% frequency range was achieved at the Naval Research Laboratory, Washington DC Studies of a number of concepts such as Free Electron Masers, Gyrotron Backward Wave Oscillators, and Cyclotron Auto-Resonant Masers towards meeting the future needs of the fusion program are also receiving support.

Fig. 3. The 400 KW, 140 GHz cw gyrotron developed by Varian.

2.3. Plasma fueling and particle control

Supplying fresh fuel to the plasma by injection of high speed solid pellets of frozen hydrogen or its isotopes has been found to improve its confinement properties. ORNL has been developing advanced pellet fueling systems that can deliver pellets with velocities up to 2 km/s using pneumatic acceleration and up to 1.5 km/s using centrifugal acceleration. While pneumatic devices produce higher velocity pellets at low repetition rates, the centrifugal system is capable of injecting a nearly unlimited number of pellets at lower speeds for continuous fueling. An eight-barrel multiple shot deuterium pellet injector was used on TFTR [8] to achieve world record plasma density ($n_e = 4 \times 10^{14}$ cm^{-3}) and a Lawson product of 1×10^{14} cm^{-3} s. ORNL also built a three-barrel pneumatic injector and installed it on the JET device. ORNL is at present building an advanced centrifugal device with continuous feuling capability for the Tore Supra Tokamak. Fig. 4 shows the Tritium Proof of Principle pneumatic injec-

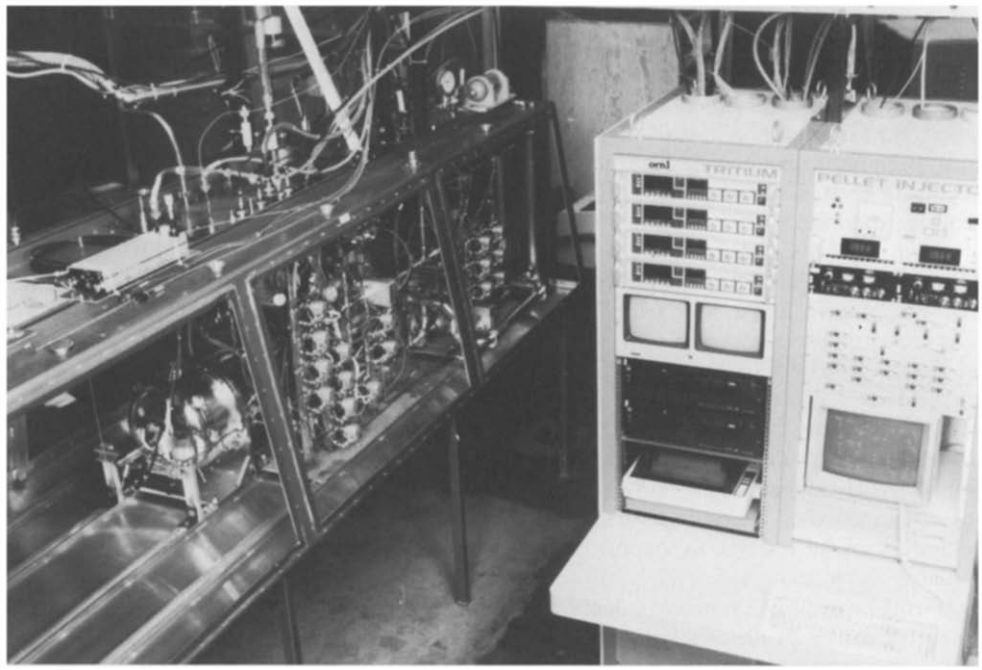

Fig. 4. The tritium proof of principle experimental facility built by ORNL at TSTA, Los Alamos National Laboratory.

tor built by ORNL, now being tested with tritium pellets at the Tritium systems Test Assembly (TSTA) at Los Alamos National Laboratory (LANL). ORNL is also engaged in investigating electron beam induced ablation techniques, two stage pneumatic principles, and arc assisted pneumatic concepts to produce higher speeds. Research is also being carried out to develop 2–5 km/s injectors that use railgun principles at the University of Illinois. Both the arc assisted pneumatic and railgun devices have achieved speeds in excess of 2 km/s with 4 mm hydrogen pellets.

Removing impurities, and particularly helium, resulting from the nuclear reaction is an equally important function in fusion devices. The vacuum system is the heart of the impurity removal program. The program has been funding the development of essential components that are not yet commercially available. These include continuously operable high vacuum cryogenic pumps, advanced seals, and improved leak detection systems. For example, the continuous duty charcoal sorption cryopump constructed by Grumman Aerospace Corporation in collaboration with LLNL has demonstrated a helium pumping rate of 7.6 $l/s/cm^2$ of panel surface area independent of gas pressure up to 10^{-4} torr [10]. The fusion program also collaborates with the National Bureau of Standards in establishing accurate methods for measuring ultra high vacuum.

3. Fusion technologies

The Fusion Technology Program supports the activities in *Fusion materials, Fusion nuclear technology,* and *Environment and safety.*

3.1. Fusion reactor materials

The ultimate economics and acceptability of fusion energy will depend to a large extent on the limitations of the materials available for constructing the various components of fusion reactors. The materials program is focused on the development of the materials science and technology base that is uniquely required for fusion devices. It is broken down into two subdivisions: *neutron interactive materials* and *plasma interactive materials.*

Neutron interactive materials

The objective of the neutron interactive materials area is the development of new or improved materials

that will enhance the economic and environmental attractiveness of fusion. This area involves the development of a fundamental understanding of the effects of the unique fusion-reactor environment on the properties and performance characteristics of materials under projected operating conditions and includes the development of baseline data on nonirradiated materials to support irradiation experiments.

In the near term (next 10 years), this element will focus on irradiation testing utilizing fission reactors and charged-particle accelerators. In the long term, utilization of a high-flux, high-energy-neutron (14-MeV) materials test facility will be required. This program also involves development of a predictive capability for relating irradiation data from available test facilities to expected performance at extended lifetimes in a fusion-reactor environment. The general long term, reactor relevant, focus of this program element has been augmented to include a focus of research in support of the ITER design study.

International collaboration is a major vehicle for accomplishing the objectives of the neutron interactive program. Long term agreements with JAERI and MONBUSHO in Japan bring significant resources and talent to ORNL and the Pacific Northwest Laboratory (PNL) for collaborative work in the High Flux Irradiation Reactor (HFIR) and the Fast Flux Test Facility/Materials Open Test Assembly (FFTF/MOTA) (see fig. 5). An expanded IEA cooperation has been initiated including collaboration in theory (supercomputer methodology), ceramics, tritium breeding solids, and the development of a common data base.

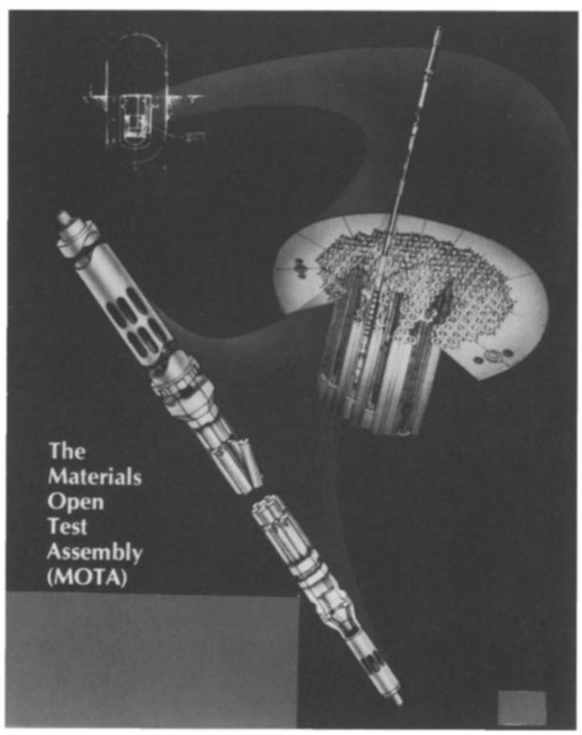

Fig. 5. The Fast Flux Test Facility – Materials Open Test Assembly located in Richland, Washington.

A new annex to the International Energy Agency Implementing Agreement on Fusion Materials has been implemented among the DOE, JAERI for Japan, and AECL for Canada for a jointly funded set of experiments of tritium breeding in the fusion sponsored FFTF/MOTA. The re-establishment of exchange activities between the US and the USSR on the subject of fusion materials that was initiated in 1987 is being extended to cooperative research with a focus on the development of reduced activation alloys and other tasks relevant to ITER.

Results of irradiation testing and metallurgical research on reduced activation compositions of ferritic and austenitic steels continue to demonstrate the validity of the approach of elemental tailoring in the development of reduced activation alloys. Special attention is now being applied to the evaluation of the potential of a family of Mn stabilized austenitic stainless steels as a candidate for application in ITER design. Selected vanadium alloys also continue to look promising in irradiation studies.

Results from the collaborative DOE/JAERI irradiation testing program in the HFIR and ORR reactors at ORNL are the only irradiation effects data on the candidate austenitic stainless steels for irradiation conditions which correspond to the low temperature range now selected as the design target for ITER. Those spectral tailoring experiments will be extended to higher fluence in a new set of irradiation tests that are scheduled to begin in October 1988.

Plasma interactive materials

The plasma interactive materials (PMI) are those located inside the vacuum vessel and subject to neutral and charged particle interactions. In the plasma interactive program, the main facilities are the Plasma Materials Test Facility (PMTF) (fig. 6) and the Tritium Plasma Experiment (TPX) at Sandia National Laboratories (SNL), and the Plasma Surface Interaction Research Facility (PISCES) at the University of California, Los Angeles (UCLA).

The Plasma Material Interaction/High Heat flux (PMI/HHF) programs are coordinated national efforts

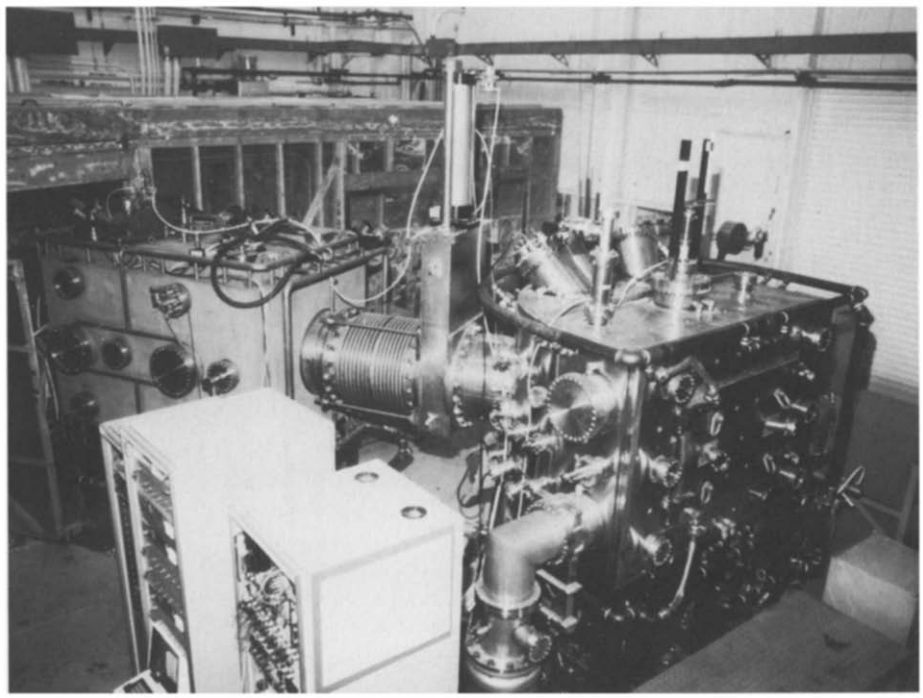

Fig. 6. PMTF-ion beam test system located at SNL.

providing data, materials, and components to such facilities as TEXTOR, Tore Supra, ASDEX, JET, TFTR and DIII-D.

An ongoing series of pumped limiter simulation experiments has been performed using the PISCES plasma. The efficiency of plasma neutralization and pumping versus variables such as pump limiter throat length, electron temperature, density, and incoming plasma flux is being investigated for both hydrogen and helium plasmas. A dramatic reduction (up to a factor of five) in the axial plasma flux as it travels down the pump limiter throat has been seen. An active limiter program has been initiated at UCLA to study the possibility of influencing the edge plasma through electrostatic limiters.

PISCES has been integrated effectively with the national program and used extensively for international collaborations with visiting scientists from IPP Garching, KFA Jülich, ENEA Frascati, IPP Nagoya, IPP Leshan (PRC) and Ben-Gurion University of the Negev.

The PMI Program has established a major collaboration with Princeton Plasma Physics Laboratory (PPPL) on issues associated with D–T operation in TFTR and CIT. All plasma-facing and neutral beam materials have been tested for tritium compatibility in the TPX. Model for tritium recycling behavior have been developed, and an in-vessel tritium inventory estimate for D–T operation in TFTR has been established. Tritium inventory estimates have been made for CIT, and collaborative measurements with INEL on tritium release rates from tritiated carbon surfaces during accidental air exposures have been initiated. The PMI properties of beryllium are being investigated in collaboration with JET, and erosion/redeposition studies are also underway in DIII-D. The HHF program has had impacts on several major facilities [11]. Extensive thermal tests have been conducted for TFTR limiters, and the PMTF was used to qualify a new first wall divertor tile design for DIII-D. A large number of long pulse, high heat flux limiter elements have been tested for Tore Supra, and a program is underway to test wall armor for the LANL and Italian reversed field pinch machines. A project for JET to design and test a pump limiter system is presently underway. A series of dump plate tests for JET are now being completed, and beryllium issues relevant to their use are being evaluated.

3.2. Fusion nuclear technology

Fusion nuclear technology, which encompasses the key fusion device components required to handle the

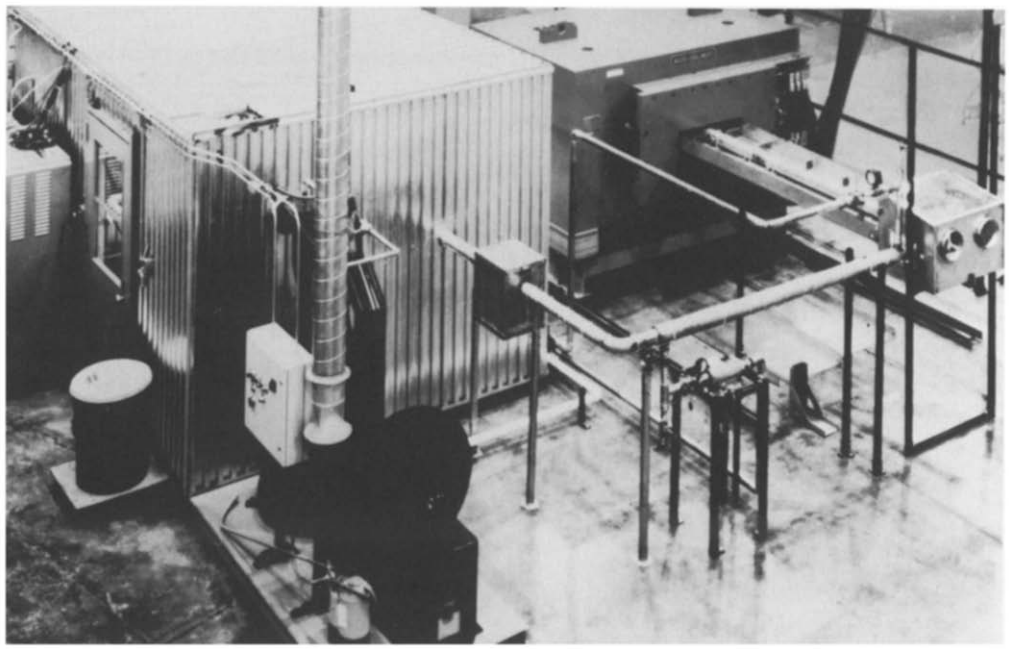

Fig. 7. The ALEX facility at Argonne National Laboratory.

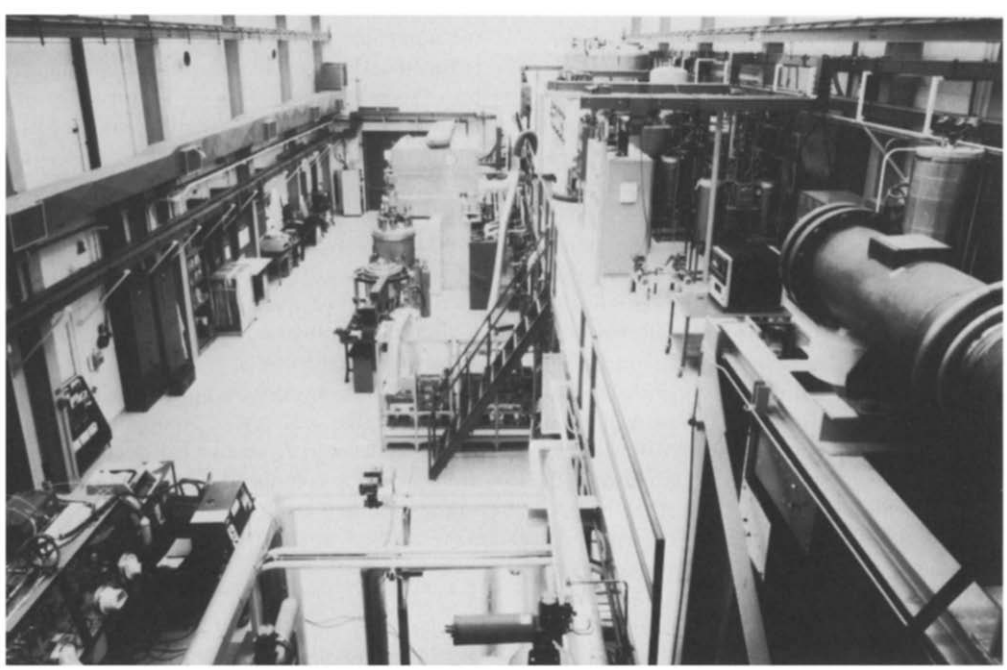

Fig. 8. The TSTA facility at Los Alamos National Laboratory.

radiation environment produced by the plasma, is one of the key technical issues for magnetic fusion. It includes *blanket technology, tritium processing technology,* and *nuclear analysis methods / data activities.*

Blankets technology

In the area of blanket technology, efforts are focused on studies of critical feasibility issues for the most promising blanket concepts. For liquid breeder blanket concepts, experimental activities are being conducted with the Argonne Liquid Metal Experiment (ALEX), shown in fig. 7, to study magnetohydrodynamic effects on pressures and flows associated with liquid metal transport through magnetic fields characteristic of fusion conditions. Data from ALEX is used to guide the development of models and computer codes for predicting the performance of self-cooled liquid breeder blankets. Experiments have been completed in ALEX on circular and rectangular cross-section flow channels as well as a more complex "flow tailoring" module in collaboration with KfK. For solid breeder blanket concepts, U.S. participation is continuing in the international BEATRIX programs under IEA auspices to develop solid breeder materials and study with fission reactors the irradiation performance of the most promising solid breeder materials. Data is rapidly accumulating on the performance of a wide range of candidate solid breeder materials tested in both open and closed capsules. BEATRIX-II will provide high-fluence, in-situ tritium recovery tests in the Fast Flux Test Reactor beginning in mid 1989.

Experiments on tritium breeding are carried out in fission reactors such as the Oak Ridge Research Reactor (ORR), High Flux Isotope Reactor (HFIR), and Fast Flux Test Facility (FFTF). Present experiments in this program area consists of static solid breeder irradiations.

Tritium processing technology

Experiments on tritium processing and handling are carried out at TSTA at LANL. Since June 1987, TSTA has been funded jointly by DOE and JAERI in a collaborative program. The facility, shown in fig. 8, is designed to test the technology required to control, separate, and purify deuterium and tritium from the plasma exhaust. TSTA contains a full tritium processing loop and has the capability of handling up to 130 grams of tritium. The facility construction was completed in 1982, and the full processing loop became operational with tritium in 1988. It has operated with up to 100 grams of tritium circulating in the system thus far [12]. Studies are presently underway with Argonne National Laboratory (ANL) to consider expanding the capability of TSTA to demonstrate the technology for processing different blanket concepts [13]. Outside of the processing system, LANL is conducting a number of tests, including testing an ORNL pellet injector [14] with tritium and testing other possible material that can be used as a storage bed for tritium. In addition, LANL is testing under Annex III two JAERI designed "process-ready" components, the palladium diffuser and the ceramic electrolysis cell. Both of these components should be a significant improvement of the present fuel cleanup system (FCU) of TSTA. These two components will be part of JAERI's newly designed and developed FCU that will be provided to TSTA by JAERI in the near future.

Nuclear analysis methods / data activities

In the area of nuclear analysis, efforts are continuing on development and dissemination of computational methods and data bases needed to support the nuclear design and analysis of fusion devices. A major element in this area is the collaboration between the U.S. and Japan on blanket integral neutronics experiments in Japan's Fusion Neutron Source facility. These experiments establish a data base for evaluation of the tritium breeding performance of blankets using lithium oxide as the breeding material. The first phase of experiments on simple blanket models has been completed; the second phase of experiments on more advanced prototypical blanket models will be completed in late 1988; and the third phase of experiments simulating a line source of neutrons will begin in late 1989 [15].

3.3 Environment and safety

The environment and safety program is responsible for assessing and evaluating the environmental and safety issues related to the operation of fusion reactors and for ensuring the development of appropriate safety technology to meet all the requirements for public and operator safety. A program is in place at the Idaho National Engineering Laboratory (INEL) to study the various aspects of fusion safety. The INEL fusion safety program can be divided into two research areas: the analytical or methodology development area and the experimental area. In the analytical area, INEL has completed the draft environmental assessment of siting the CIT at Princeton. Most of the analytical effort has been concentrated on the development of codes and models which can demonstrate the operational and safety aspects of fusion systems under different conditions. It will be important for fusion to develop and demonstrate its safety features to help ensure public

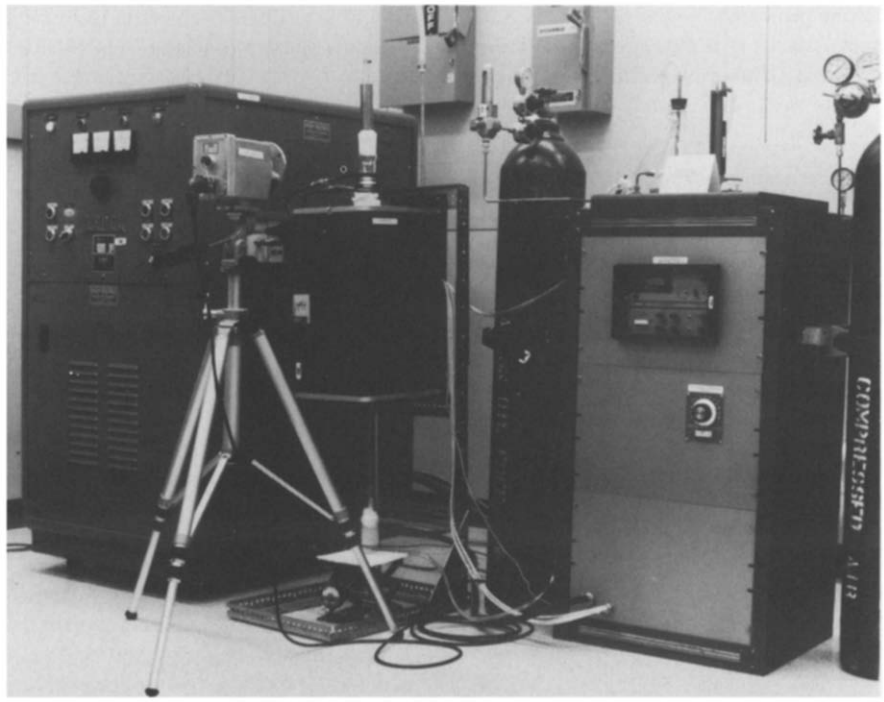

Fig. 9. The experimental set up to study release of activation products at INEL.

and operator safety as well as public acceptance. In addition, a study (ESECOM) has been completed that examined the relationship between environment, safety, and economics (ESE) to assess magnetic fusion energy's potential in these three areas. The summary of the ESECOM report (UCRL-53766 Summary) was completed by LLNL in 1987 [16].

In the experimental area, the three major efforts over the past year have been in tritium, activation product behavior, and liquid lithium safety. In the tritium area, INEL is continuing its experiments in implantation/permeation of plasma facing materials to determine the possible extent of tritium build-up and release mechanisms in the material. The tests this year have focused on a carbonaceous coating. In the activation product behavior area, INEL is studying release of activation products by heating potential fusion materials to high temperatures and measuring the volatized elements. The experimental setup is shown in fig. 9. Materials that have been studied include austenitic steels (PCA), ferritic steels (HT-9), austenitic manganese-stabilized steels (AMCR-033) and vanadium alloys ($V_{15}Cr_5Ti$) [17]. The major work this year will be on tungsten alloy mobilization tests in air and in steam. In the liquid lithium safety area, HEDL is continuing with its tests of liquid lithium in different environments. The tests this year have concentrated on liquid lithium pool at high temperatures (700 °C) in an atmosphere of steam and argon and a high temperature lithium spray (750 °C) in a steam atmosphere. These tests give valuable data on potential accident scenarios and the consequences of utilizing liquid lithium in a fusion reactor.

4. Fusion design studies

The work funded by the Fusion design Studies Program falls into two categories. The near-term experimental device studies have included the CIT and currently include the ITER study that is being pursued along with the European Community, Japan, and the Union of Soviet Socialist Republics. The long-term studies of commercial electricity generating devices are represented by the completed TITAN Reverse Field Pinch Reactor Study and the ongoing Advanced Reactor Innovations Evaluation Study (ARIES).

The major participants in these study efforts include personnel from ANL, UCLA, PPPL, LLNL, University of Wisconsin, Rensselaer Polytechnic Institute, University of Illinois, ORNL, MIT, and the FEDC, as well as

industrial support provided by Grumman Aerospace Corporation, GA Technologies, TRW, and McDonnell Douglas. The focus of near-term experimental device studies since the last Symposium on Fusion Technology has been in the area of a nuclear test facility that has the capability to address controlled ignition and extended burn of deuterium–tritium plasma. In the U.S., the TIBER study proposed a design for such a device. The design used advanced engineering concepts resulting in a compact, low cost device that is reactor relevant. Important features of the design include radiation resistant, high critical current density toroidal field magnets, minimum neutron shielding, compact divertor, and compact structures, all of which lead to small overall size and, therefore, low cost. The capability to operate steady state with a high neutron wall loading and a modest availability results in a relatively high neutron fluence at the nuclear testing positions of the device. This design formed the basis for the U.S. proposals to the ITER design process that began in January 1988 and will complete its initial work in December 1990 by producing a Conceptual Design Report for a device that can address both reactor relevant physics issues as well as critical issues for fusion nuclear technology.

The TITAN RFP Reactor Study was completed in December 1987. The study lead by UCLA, included participation from GA Technologies, RPI, LANL, and FEDC. Culham Laboratory in the UK provided one person full time throughout the project to work at UCLA while two workshops on RFP design were held with the Japanese during the course of the study. Two versions of the TITAN reactor design were produced: a liquid lithium cooled, vanadium alloy structure material blanket design and an aqueous loop-in-pool concept with dissolved Li salt as the breeder and a reduced activation ferritic steel as the structural material. Both designs showed that high neutron wall loading (10–20 MW/m^2), compact, high power density (500–700 We/tonne) fusion systems can be technically feasible and economically attractive. Each design permitted single-piece maintenance of the fusion power core and satisfied current U.S. NRC regulations for near-surface burial of radioactive wastes. The study showed that compact reactors enhanced economics but did not imply increased safety or environmental problems.

The Advanced Reactor Innovations Evaluation Study (ARIES) is a three year effort that began in January 1988. It will explore several "visions" of what an attractive tokamak power reactor may look like given the physics and technology advances that have been made since the completion of the STARFIRE study eight years ago. During this first year of the study, the team, along with help from Culham Laboratory and Japan, is exploring many methods for improving the tokamak concept. Possibilities include high $\beta(>20\%)$, direct conversion, very low activation materials, advanced fuels, low aspect ratio (~ 1), high aspect ratio (~ 6), pool designs, room temperature superconducting magnets, and so on. At the end of this year all of the individual ideas will be assessed and combined into two or three self consistent visions that will be studied in greater detail during 1989 and 1990.

Conclusion

Much progress has been made in the last two years in the U.S. magnetic fusion technology program. Notable advancements have been made in superconducting magnets, gyrotrons, reduced activation materials, pellet injectors, and safe handling of tritium to name a few.

Advancements in fusion plasma conditions and economic, environmentally attractive reactor designs will continue to require significant improvements in the technology of magnets, heating, fueling, materials, and safety. The ITER process will provide a forum and an opportunity to accelerate the development of fusion technology through new international collaborations, and we should take advantage of that opportunity.

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