PREPARED FOR THE U.S. DEPARTMENT OF ENERGY,
UNDER CONTRACT DE-AC02-76CH03073

PPPL-3630
UC-70

PPPL-3630

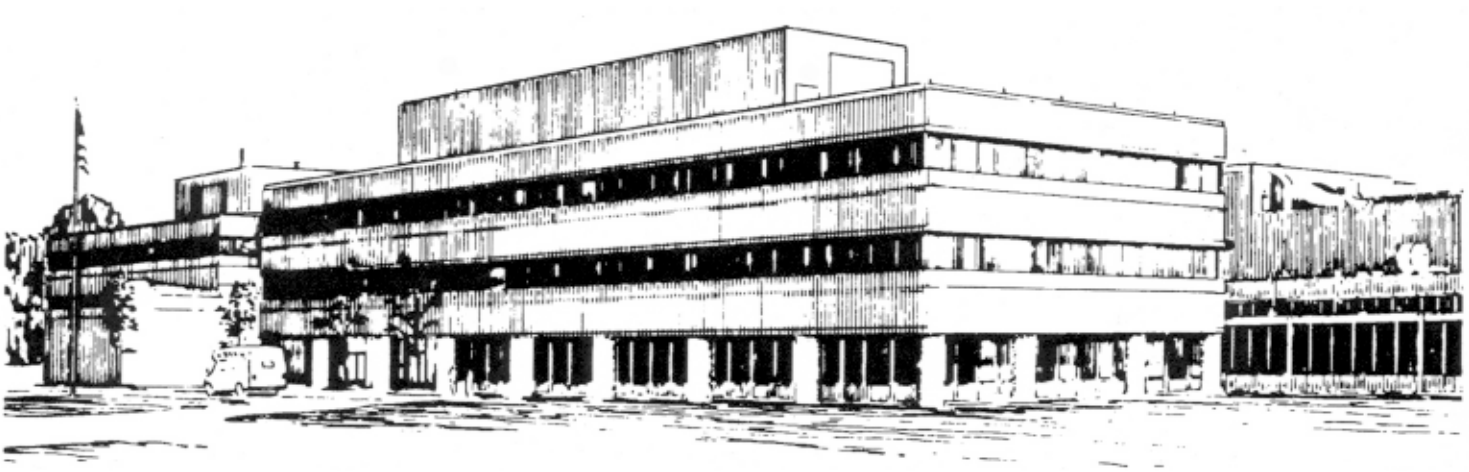
Tritium Removal by Laser Heating and Its Application to Tokamaks

by

C.H. Skinner, C.A. Gentile, G. Guttadora, A. Carpe, S. Langish,
K.M Young, M. Nishi, and W. Shu

November 2001

PPPL
PRINCETON PLASMA PHYSICS LABORATORY



PRINCETON PLASMA PHYSICS LABORATORY
PRINCETON UNIVERSITY, PRINCETON, NEW JERSEY

PPPL Reports Disclaimer

Availability

This report is posted on the U.S. Department of Energy's Princeton Plasma Physics Laboratory Publications and Reports web site in Fiscal Year 2002. The home page for PPPL Reports and Publications is: http://www.pppl.gov/pub_report/

DOE and DOE Contractors can obtain copies of this report from:

U.S. Department of Energy
Office of Scientific and Technical Information
DOE Technical Information Services (DTIS)
P.O. Box 62
Oak Ridge, TN 37831

Telephone: (865) 576-8401

Fax: (865) 576-5728

Email: reports@adonis.osti.gov

This report is available to the general public from:

National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

Telephone: 1-800-553-6847 or
(703) 605-6000

Fax: (703) 321-8547

Internet: http://www.ntis.gov/ordering.htm

TRITIUM REMOVAL BY LASER HEATING AND ITS APPLICATION TO TOKAMAKS

C.H. Skinner, C. A. Gentile, G. Guttadora,
A. Carpe, S. Langish, K. M. Young
Princeton Plasma Physics Lab,
Princeton
NJ 08543, USA

M. Nishi, W. Shu
Tritium Engineering Laboratory,
JAERI, Ibaraki,
Japan

ABSTRACT

A novel laser heating technique has recently been applied to removing tritium from carbon tiles that had been exposed to deuterium-tritium plasmas in the Tokamak Test Fusion Reactor (TFTR). A continuous wave neodymium laser, of power up to 300 watts, was used to heat the surface of the tiles. The beam was focussed to an intensity, typically 8 kW/cm^2, and rapidly scanned over the tile surface by galvanometer driven scanning mirrors. Under the laser irradiation, the surface temperature increased dramatically, and temperatures up to 2,300 C were recorded by an optical pyrometer. Tritium was released and circulated in a closed loop system to an ionization chamber that measured the tritium concentration. Most of the tritium (up to 84%) could be released by the laser scan. This technique appears promising for tritium removal in a next step DT device as it avoids oxidation, the associated deconditioning of the plasma facing surfaces, and the expense of processing large quantities of tritium oxide. Some engineering aspects of the implementation of this method in a next step fusion device will be discussed.

1. INTRODUCTION

Tritium removal from plasma facing components is arguably the most serious challenge facing next step magnetic fusion devices that use carbon plasma facing components. The long-term tritium inventory for ITER is limited to about 350˚g, mainly due to safety considerations. It is appears likely that this inventory limit will be reached after about 100 pulses,[1] requiring tritium removal by methods yet to be established, before plasma operations can resume. Independent of safety considerations, tritium is expensive, the supplies are limited and hold-up of large quantities of tritium on the vessel walls is impractical for DT fusion reactors.

The dominant mechanism for tritium retention in fusion devices with carbon plasma facing components is codeposition. Carbon is eroded by plasma ions and neutrals and is redeposited together with tritium in co-deposited layers on plasma facing and hidden surfaces. A high fraction of the tritium injected into TFTR and JET (the Joint European Torus) was retained in the vessel.[2,3] Techniques for tritium removal have been demonstrated in the laboratory, and on tokamaks[4-6] but they are relatively slow. For example, several weeks were devoted to tritium removal from the TFTR vessel after a cumulative 10-15 minutes of DT plasma operations. Clearly for a practical fusion reactor, the time devoted to tritium removal should be much less than that for plasma operations. Present candidate detritiation techniques generally involve oxidation which deconditions the vessel walls and requires additional time devoted to removing oxygen from the surface to restore good plasma operations. Oxidation also generates undesirably large quantities of DTO exhaust that will be expensive to reprocess.[7] Tritium issues are arguably the most sensitive area of fusion for the public and demonstration of suitable tritium removal technology is urgently needed.

Recently a novel method of removing tritium by laser heating was proposed.[8] It is well know that tritium is released from graphite at high temperatures, but heating the whole vacuum vessel to the required temperatures is impractical. On the other hand most of the tritium is bound up in surface layers that are of order 50-100 microns thick in contemporary tokamaks. These codeposited layers can be heated by a scanning laser beam. Heat transfer calculations showed that a flux of about 3 kW/cm^2 could heat a graphite surface to 2300 C in about 20 ms, and the heat pulse would penetrate through a 50-100 micron depth that matched the thickness of the codeposits.

2. EXPERIMENTAL RESULTS

A tabletop experiment was set up to study this scheme. To date 16 samples have been irradiated in experiments that varied the laser intensity, focal spot size, scan speed, atmosphere in the chamber (argon or air) and the results[9] are summarized below. Graphite and carbon fiber composite (CFC) tiles that had been exposed to DT plasmas in TFTR were retrieved from the vacuum vessel and cut into cubes of 2-cm linear dimension. Fig. 1 shows the plasma facing surface of a cube that showed an interesting variation in surface texture. The cubes were positioned inside a vacuum chamber with two access windows (Fig. 2). A 300 watt continuous wave neodymium laser beam was directed by two galvanometer driven scanning mirrors through a lens that focussed it on the surface of the cube. The size of the focal spot typically used was 2 mm and focal intensity 8 kW/cm^2. The mirrors could be rapidly swiveled, scanning the focal spot on the cube surface at speeds up to 2 m/s over a 75 x 75 mm field. The trajectory of the focal spot on the tile surface was pre-programmed with a PC computer that controlled the mirrors. Typically, a raster pattern with line spacing 0.5 mm was used.

A pyrometer, operating in the wavelength range 1.58 — 1.8 microns, measured the surface temperature rise. The pyrometer viewing area was 0.7 mm diameter, the response time 0.3 msec and the temperature data was digitized and acquired by the PC. A filter blocked scattered Nd laser radiation from the pyrometer. A given location on the cube surface experienced first the fringe of the laser spot, then the peak intensity, then the fringe in successive passes (Fig.3). The temperature of the codeposits increased dramatically with the emission of brilliant incandescence and the peak temperature exceeded 2,300 C. A digital microscope was used to image the tile surface before, during and after the laser irradiation. The atmosphere in the chamber was circulated in a closed loop system through an ionization chamber that measured the tritium concentration. The tritium released was then calculated from the concentration and the system volume.

The temperature rise was found to be strongly dependent on the surface material and was much higher for a codeposit than the manufactured tile material (graphite or CFC). Fig. 4 compares the temperature excursion of the codeposition area (A) and erosion area (B) of Fig. 2. The temperature of the codeposited surface rose to 1841 C while the erosion area increased to only 1141 C for the same laser intensity. The more open porous structure of the codeposits leads to a reduced thermal conductivity and enhanced surface temperature rise, which is advantageous for releasing tritium.

Fig. 1 Microscope image of KC17-2B before laser irradiation showing an erosion region (A) and codeposition region (B). The area shown is 10 mm wide.

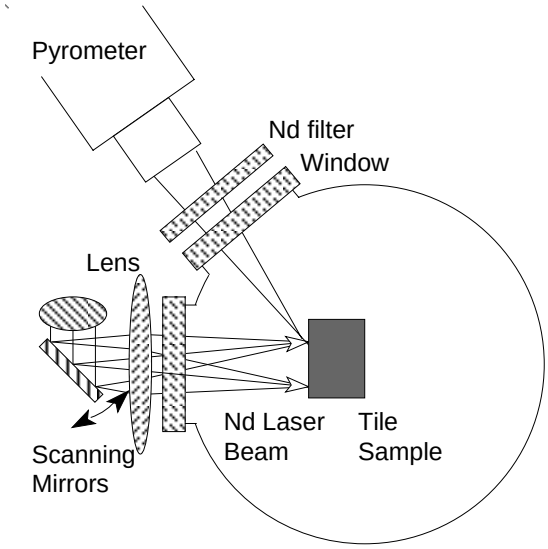


Fig. 2 Experimental setup

The digital microscope was used in video mode to image the incandescence (Fig. 5). The laser power was 8 kW/cm^2, atmosphere argon, and the scan speed 50 mm/s. The light emission was attenuated by x100,000 with a neutral density filter. The image revealed the presence of 100-micron scale hot spots . Images of codeposits [10] show grains and pores with a 10-100 micron scale and it is clear that thermal conduction will also vary strongly on this spatial scale leading to strong gradients in surface temperature.

Tritium release was measured by an ionization chamber. Fig. 6 shows the rise in the total tritium from one scan. Note the tritium is not reabsorbed after the scan. The quantity of tritium released was then calculated from the

concentration and the system volume. Strong tritium release was observed at surface temperatures above 1,230 C. An ambient atmosphere of air or argon was used in the chamber and had no measurable effect on the amount of tritium released, though in the presence of air most of the tritium was converted to the oxide form (as measured by a differential tritium sampler).

The fraction of the total tritium released by the laser scan is a key parameter. It is well known that baking codeposits in air at temperatures of 350 C releases nearly all the tritium.[11] Following the laser scan, an unfocussed stationary laser spot was used to heat the cube to > 350 C in an air atmosphere for 40 minutes releasing nearly all of the remaining tritium. The fraction of the total tritium that was released by the laser scan was 46% and 84% in two experiments at 8 kW/cm^2 and 31 kW/cm^2 focal intensity.

Ablation of surface material is undesirable as the tritiated debris may be redeposited in hidden areas and be difficult to retrieve. At high scan speeds (1 m/s) and 8 kW/cm^2 intensity there was minimal change in the surface appearance and analysis of debris from the inside of the chamber showed 99.9% of tritium was released in gaseous form. The surface temperature rise and tritium release did not increase appreciably at lower scan speeds, however the deeper penetration of the heat pulse beyond the codeposited layer led to significant surface damage at speeds of 0.2 m/s and below. The duration of the temperature excursion at 1 m/s is 10-20 ms and the depth of heat penetration predicted by modeling a good match to the 50-100 micron thickness of the codeposited layer.

3. IMPLEMENTATION IN A TOKAMAK

In this section we consider some of the engineering issues associated with implementation of this technique in an operating tokamak. The goal is to design a system for possible use on a next step DT tokamak with carbon plasma facing components, such as ITER. Almost every other aspect of ITER has a detailed predictive knowledge base, from decades of tokamak operations and dedicated full—scale engineering demonstrations (e.g. of remote handling technology). Near term demonstration of rapid and efficient tritium removal on a contemporary tokamak such as JET is urgently needed.

The primary requirement is to deliver enough energy to heat a 100 m surface layer to 2000 C. In ITER the area of carbon tiles is 50 m^2 and the energy required is 3×10^7 J. Multi-kW continuous power is available from diode, Nd and CO_2 lasers and the required energy could be delivered by a 3kW laser over 3 hours. The laser beam will be transmitted to the scan head inside the vessel via fiberoptics. CO_2 lasers, while very efficient, have the disadvantage that the beam is not transmissible by fiber

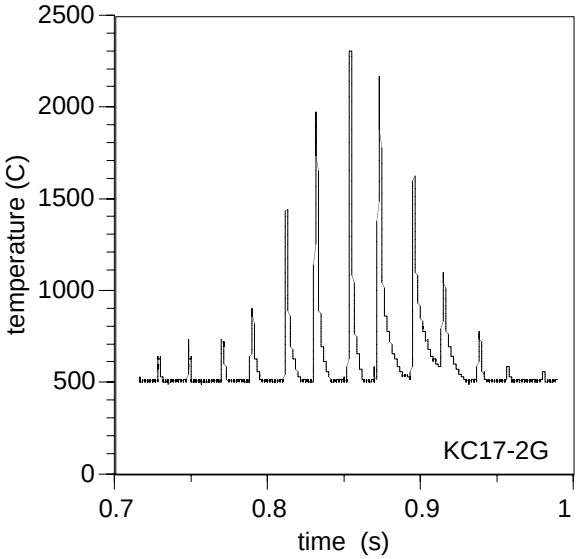


Fig. 3 Temperature time history as the laser rasters over the point viewed by the pyrometer successive times. The pyrometer operating range is 500 — 2300 C.

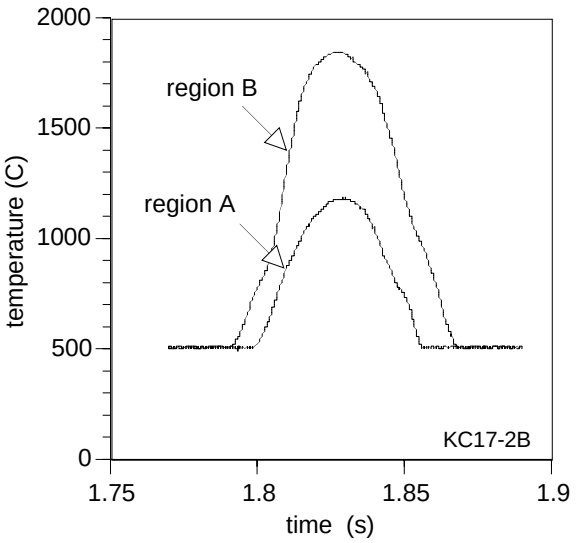


Fig. 4 The peak temperature excursion of erosion areas (A) and codeposition area (B) of Fig. 2 with laser intensity 8 kW/cm^2, scan speed 50 m/s.

optics and so will not be considered further. Diode lasers have powers up to 3kW. These are more efficient than Nd lasers however their higher divergence leads to a larger diameter coupling fiber and lower flux. Industrial fiber coupled Nd lasers are available up to 6kW multimode[12] and are the lead candidate. The anticipated co-deposition rate in ITER is 1-10 nm/s so that with a 10% duty cycle the codeposit thickness will be 60 - 600 microns after one week. Weekly detritiation is envisaged.

Fig. 5 Still image from a video taken of the Nd laser interacting with the surface of KC17 cube 2B. The image covers 4.5 mm in the horizontal direction. Micro hot spots are evident.

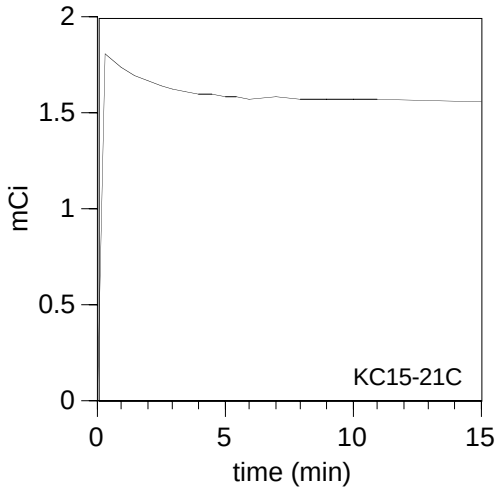


Fig. 6 Tritium release measured by ionization chamber.

Figure 3 reflects the intensity distribution in the present laser focal spot, averaged over the 0.7 mm pyrometer area. Clearly a top hat uniform focal spot with a well defined area of uniform intensity would be advantageous and enable the laser power to be tightly optimized. This profile arises naturally from imaging a fiber optic delivered beam. Experience so far has shown that the surface temperature rise is variable and a strong function of the codeposit thermal conductivity. Feedback control of the laser power to attain a specified temperature rise would be advantageous. Dichroic mirrors could provide a co-axial sight line of the pyrometer along the laser axis. The temperature of the laser spot on the tile surface would then be measured continuously and input to a feedback loop that controlled the laser power. Rapid control of the laser power could be provided by electro optical means such as a Pockel cell. These are typically used within the laser cavity, but when opened, the stored energy in the laser cavity generates an initial burst of power. For this reason it may be advantageous to use a Pockel cell external to the laser cavity that diverted some of the laser power to a beam dump.

The fiber optic and scanner will be entering a harsh radiation environment. After 20 years of operation the radiation level inside the ITER vessel due to gammas from activation is estimated to be 10,000, 2,000, 500 Gy/hr at a time 10^4, 10^5, 10^6 s after shutdown.[13] Recently developed radiation resistant optical fibers have showed low radiation induced loss.[14] It appears that the loss will be 0.1-0.5 dB/km/kGy at the most[15] which would be tolerable.

The working distance from the scanning lens to the tile surface is constrained by the desired laser flux on the surface. A 1˚kW laser beam transmitted by a 600 micron fiber can provide a flux of about 20 kW/cm^2 on target using a scanner with working distance of 35 cm. Sufficient flux cannot be delivered to the divertor region by a scan head positioned much further away e.g. from a centrally positioned periscope. A remotely controlled vehicle, such as was vividly demonstrated by the Sojourner expedition to Mars, could carry the scanner and diagnostics. The magnetic field at the ITER inner divertor is about 6.1 T. While demagnetization can be performed in a few hours, the superconducting TF magnets are limited to about 1000 operational transitions. It will be necessary for the in-vessel hardware to be made from non-magnetic materials and be capable of functioning in a high magnetic field. Piezo-electrically actuated scanning mirrors could replace the present electromagnetic galvanometer drivers.

The present ITER divertor (Fig. 7) is designed to optimize power and particle handling rather than provide laser access to the private region under the dome or into the small (0.3 mm) gaps in the tiles. If laser detritiation is adopted then some modifications to the divertor configuration may be desirable.

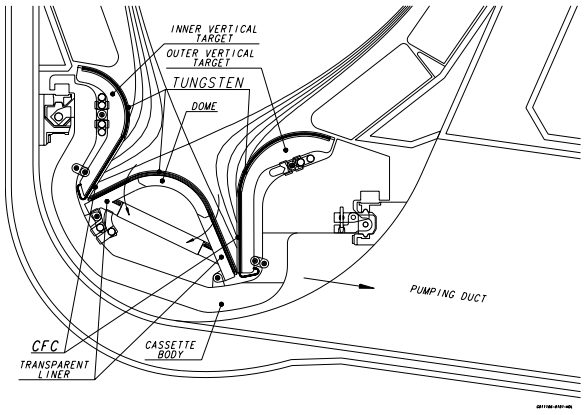


Fig. 7 The present design of the ITER divertor.

4. FUTURE WORK

Experiments are planned to further study some issues. Fiber optic coupling of the laser to the scan head will be implemented and the optimal laser intensity for maximum tritium release with minimal change to the surface morphology will be measured. Diffusion of tritium deeper into the tile is unlikely during the millisecond duration of the heat pulse, but this will be checked by accelerator mass spectrometry of laser treated samples at the Tritium Laboratory Karlsruhe.

Demonstration of this technique in a contemporary tokamak is needed. Carbon tiles in operating tokamaks are extensively conditioned to minimize trapped oxygen (water) on the surface. Redeposition of the released tritium either back onto the tile, or to the chamber walls was not observed in the present experiments (Fig. 6) but checks in contemporary tokamaks are desirable. A near term experiment with a scanner module transported by the remote manipulator arm in JET would be an important step in the development of this technique. The arm would be positioned to sequentially scan the tiles in the divertor region. Released tritium would be detected by a sniffer system with an ionization chamber. The detritiation factor would be assessed by subsequent ex-vessel analysis of tiles.

During previous JET DTE-1 operations most of the tritium inventory was trapped on flakes in hard-to-access regions in the sub divertor. Access to this region by an articulated fiber optic is possible and miniaturized scanners could be developed to detritiate this region. During JET maintenance periods the magnetic field is off and the vessel at atmospheric pressure. In one design concept the focussed Nd laser beam emerges via a miniature rotating scan head that produces a circular region of intense heating that will thermally release tritium from debris and flakes. The combination of rotary motion (from the scan head) and axial motion (by moving the fiber) will allow coverage of hard-to-access regions. Prototype development is needed to validate this concept.

ACKNOWLEDGEMENTS

We wish to acknowledge stimulating exchanges with A. Costley, G. Federici, A. Rolfe, and M. Schimpf. Support was provided by the Annex IV to the JAERI/DOE Implementing Arrangement on Cooperation in Fusion Research and Development, U.S. DOE Contract Nos. DE-AC02-76CH03073

REFERENCES

[1] G. Federici et al., "Assessment of erosion and tritium codeposition in ITER-FEAT" *J. Nucl. Mater.*, 290-293, 260-265 (2001).

[2] C. H. Skinner et al., "Modeling of tritium retention in TFTR" *J. Nucl. Mater.*, 266-269, 940-946, (1999).

[3] P. Andrew et al., "Tritium retention and clean-up in JET" *Fus. Eng. & Des.*, 47, 233-245 (1999).

[4] G. Federici, C. H. Skinner et al., "Plasma-Material Interactions in Current Tokamaks and their Implications for Next-Step Fusion Reactors." *Nucl. Fus.* in press.

[5] G. Federici et al., "In-vessel Tritium Retention and Removal in ITER-FEAT" *Physica Scripta* T91: 76-83 (2001)

[6] G. F. Counsell and C. H. Wu, "In-situ detection and removal of carbon debris - a challenge for the next-step fusion device" *Physica Scripta,* T91, 70-75, (2001).

[7] C. H. Skinner et al., "Tritium experience in large tokamaks: Application to ITER" *Nucl., Fus.*, 39 (2), 271-291, (1999).

[8] C. H. Skinner et al., "Tritium Removal by CO_2 Laser Heating" *Proceedings of SOFE'97 17th IEEE/NPSS Symposium on Fusion Engineering*, Vol. 1, p. 321-324, IEEE Inc. Piscataway, NJ, (1997).

[9] C. H. Skinner et al., "Tritium removal from Codeposits on Carbon Tiles by a Scanning Laser" submitted to *J. Nucl. Mater.* (2001).

[10] B. Mills, D.A. Buchenauer, A. E. Pontau, and M. Ulrickson, "Characterization of deposition and erosion on the TFTR bumper limiter and wall" *J. Nucl. Mater.,* 162-164, p.343-349 (1989).

[11] J. W. Davis and A. A. Haasz, "Oxygen removal of codeposited a-C:D layers from tokamak tiles" *J. Nucl. Mater.*, 266-269, p.478-484, (1999).

[12] e.g. Rofin Sinar Inc.

[13] H. Iida, personal communication.

[14] T. Shikama et al., "Behavior of developed radiation-resistant silica-core optical fibers under fission reactor irradiation. *Fus. Eng. & Des.*, 51-52 p 1790-183 (2000).

[15] T. Shikama, personal communication.

External Distribution

Plasma Research Laboratory, Australian National University, Australia
Professor I.R. Jones, Flinders University, Australia
Professor João Canalle, Instituto de Fisica DEQ/IF - UERJ, Brazil
Mr. Gerson O. Ludwig, Instituto Nacional de Pesquisas, Brazil
Dr. P.H. Sakanaka, Instituto Fisica, Brazil
The Librarian, Culham Laboratory, England
Library, R61, Rutherford Appleton Laboratory, England
Mrs. S.A. Hutchinson, JET Library, England
Professor M.N. Bussac, Ecole Polytechnique, France
Librarian, Max-Planck-Institut für Plasmaphysik, Germany
Jolan Moldvai, Reports Library, MTA KFKI-ATKI, Hungary
Dr. P. Kaw, Institute for Plasma Research, India
Ms. P.J. Pathak, Librarian, Insitute for Plasma Research, India
Ms. Clelia De Palo, Associazione EURATOM-ENEA, Italy
Dr. G. Grosso, Instituto di Fisica del Plasma, Italy
Librarian, Naka Fusion Research Establishment, JAERI, Japan
Library, Plasma Physics Laboratory, Kyoto University, Japan
Research Information Center, National Institute for Fusion Science, Japan
Dr. O. Mitarai, Kyushu Tokai University, Japan
Library, Academia Sinica, Institute of Plasma Physics, People's Republic of China
Shih-Tung Tsai, Institute of Physics, Chinese Academy of Sciences, People's Republic of China
Dr. S. Mirnov, TRINITI, Troitsk, Russian Federation, Russia
Dr. V.S. Strelkov, Kurchatov Institute, Russian Federation, Russia
Professor Peter Lukac, Katedra Fyziky Plazmy MFF UK, Mlynska dolina F-2, Komenskeho Univerzita, SK-842 15 Bratislava, Slovakia
Dr. G.S. Lee, Korea Basic Science Institute, South Korea
Mr. Dennis Bruggink, Fusion Library, University of Wisconsin, USA
Institute for Plasma Research, University of Maryland, USA
Librarian, Fusion Energy Division, Oak Ridge National Laboratory, USA
Librarian, Institute of Fusion Studies, University of Texas, USA
Librarian, Magnetic Fusion Program, Lawrence Livermore National Laboratory, USA
Library, General Atomics, USA
Plasma Physics Group, Fusion Energy Research Program, University of California at San Diego, USA
Plasma Physics Library, Columbia University, USA
Alkesh Punjabi, Center for Fusion Research and Training, Hampton University, USA
Dr. W.M. Stacey, Fusion Research Center, Georgia Institute of Technology, USA
Dr. John Willis, U.S. Department of Energy, Office of Fusion Energy Sciences, USA
Mr. Paul H. Wright, Indianapolis, Indiana, USA

The Princeton Plasma Physics Laboratory is operated
by Princeton University under contract
with the U.S. Department of Energy.

Information Services
Princeton Plasma Physics Laboratory
P.O. Box 451
Princeton, NJ 08543

Phone: 609-243-2750
Fax: 609-243-2751
e-mail: pppl_info@pppl.gov
Internet Address: http://www.pppl.gov