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Title: **USING THE TRITIUM PLASMA EXPERIMENT TO EVALUATE ITER PFC SAFETY[a]**

Author(s): Glen R. Longhurst and Robert A. Anderl
Idaho National Engineering Laboratory

Rion A. Causey
Sandia National Laboratories

John R. Bartlit
Los Alamos National Laboratory

John R. Haines
MDC Aerospace

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Using the Tritium Plasma Experiment to Evaluate ITER PFC Safety[a]

Glen R. Longhurst and Robert A. Anderl
Idaho National Engineering Laboratory
Idaho Falls, ID 83415-3880

John R. Bartlit
Los Alamos National Laboratory
Los Alamos, NM 87545

Rion A. Causey
Sandia National Laboratories
Livermore, CA 94551-0969

John R. Haines
MDC Aerospace
St. Louis, MO 63166

ABSTRACT

The Tritium Plasma Experiment was assembled at Sandia National Laboratories, Livermore to investigate interactions between dense plasmas at low energies and plasma-facing component materials. This apparatus has the unique capability of replicating plasma conditions in a tokamak divertor with particle flux densities of 2×10^{19} ions/cm^2·s and a plasma temperature of about 15 eV using a plasma that includes tritium. With the closure of the Tritium Research Laboratory at Livermore, the experiment was moved to the Tritium Systems Test Assembly facility at Los Alamos National Laboratory. An experimental program has been initiated there using the Tritium Plasma Experiment to examine safety issues related to tritium in plasma-facing components, particularly the ITER divertor. Those issues include tritium retention and release characteristics, tritium permeation rates and transient times to coolant streams, surface modification and erosion by the plasma, the effects of thermal loads and cycling, and particulate production. A considerable lack of data exists in these areas for many of the materials, especially beryllium, being considered for use in ITER. Not only will basic material behavior with respect to safety issues in the divertor environment be examined, but innovative techniques for optimizing performance with respect to tritium safety by material modification and process control will be investigated. Supplementary experiments will be carried out at the Idaho National Engineering Laboratory and Sandia National Laboratory to expand and clarify results obtained on the Tritium Plasma Experiment. An industrial consortium led by McDonnell Douglas will design and fabricate the test fixtures.

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INTRODUCTION

Safety is one of many issues facing the designers of plasma-facing components (PFCs) of the International Thermonuclear Experimental Reactor (ITER) and other developmental fusion machines. In many areas, the safety of systems and components may be reasonably well estimated from existing test data or using existing analytical tools. For PFCs, however, there are safety concerns for which there are few if any prior results on which to base design decisions.

The ability to simulate the environment encountered by PFCs is a key one in developing the experimental data base for PFC safety. That capability exists in the Tritium Plasma Experiment (TPE)[b], an apparatus initially constructed at Sandia National Laboratories, Livermore (SNLL), and now planned for operation at the Tritium Systems Test Assembly (TSTA) at Los Alamos National Laboratory (LANL). The experimental apparatus has a nominal particle flux of 2×10^{24} ions/m^2.s, a background gas pressure of 13 mPa, a plasma density of 2×10^{19} ion/m^3, an electron temperature of 15 eV, and a magnetic field intensity of 0.25 tesla. The volume available for the test article is nominally two liters. The configuration of the device is essentially the same as that of the PISCES-A facility at UCLA.[1]

A project involving the collaborative efforts of the Idaho National Engineering Laboratory (INEL), SNLL, TSTA, and an industrial consortium led by MDC Aerospace has been initiated to investigate safety issues for PFCs using the TPE facility. The purpose of this report is to lay the background for undertaking that task, to outline technical and programmatic ~~accomplishments~~ the project should pursue, and to provide general guidance to experiment designers. goals

The INEL has varied interests in this experimental program. The Fusion Safety Program has responsibility for safety in most aspects of design, including plasma-facing components. Seeking to assure the success of ITER, there is a strong desire to establish/improve lines of communication with other elements of the ITER Team and other fusion developers. Information needs to be gathered to be used as a basis for preparing guidance to designers and regulators. Small, supplementary experiments to augment what will be done on this project can also be performed at the INEL. SNLL has interests also. Because the TPE was developed and assembled at SNLL, the only real expertise in its operation is presently there. Additionally, SNLL research specialists in plasma material interactions will be able to contribute substantially to the understanding of experimental results obtained with TPE. TSTA contributes the laboratory space and the unique tritium handling support systems that will allow TPE to operate. The TSTA staff will also become system operators. The MDC Aerospace team has been awarded a contract by the U.S. Department of Energy (DOE) to provide engineering support services to ITER in the area

[b]Previously referred to as "TPX", the TPE designation is adopted here to avoid confusion with the Toroidal Plasma Experiment (TPX) being designed at Princeton Plasma Physics Laboratory.

of PFCs. In that role, they will be designing and developing many of the internal components for the U.S. ITER Home Team. This combination of resources should assure success of the project.

SAFETY CONSIDERATIONS

In this section, the background is laid for investigation of the various safety issues associated with PFCs. Safety is a major consideration in the ultimate viability of fusion power as an energy source. In a typical commercial fission reactor power plant in the U.S. almost half of the total operating costs are not associated with plant maintenance or fuel, but with operating procedures that are strongly driven by the costs of securing and maintaining public acceptance through documentation of safety-related performance. Approximately 20,000 pages of licensing correspondence must be prepared each year for a representative pressurized water reactor. Unless fusion can be shown to have substantially greater inherent safety characteristics that fission reactor systems, we may expect that the costs of documenting compliance with the various environmental and safety regulations will be as high for fusion as they are for fission. Since initial capital costs will be higher for fusion, there will be no incentive for commercialization. Thus, safety in fusion is one of the high-priority concerns in fusion development. Safety of plasma-facing components is a significant part of that concern.

Tritium Inventory and Accountability

On-site tritium is a major factor in the radiological hazard associated with any tritium-burning fusion reactor. While systems such as the fuel storage and injection systems, the exhaust gas cleanup system, isotopic separation, and waste processing systems will house much of the tritium inventory for the facility, large amounts may also be expected to coat the walls of beamlines and cryopumping surfaces (if used) and to reside on PFC's. In the event of a structural failure or other accident resulting in an over-temperature transient, tritium on surfaces may be released, causing a threat to workers and to the environment and its residents. That is a major reason that the amount of tritium that is authorized to be on site during operations is strictly limited.

To be in compliance with that limitation, fusion machine owners and operators must be able to determine how the tritium inventory is distributed within the fusion plant. Specifically, that includes determination of how much tritium resides in PFC's and how vulnerable that inventory is to release. That calls both for understanding the processes taking place there and for measurement methods that can confirm the presence or absence of such an inventory.

Studies have been performed to evaluate tritium uptake characteristics for certain materials such as graphite[illegible] but for others, little work has been done. Beryllium is a material being strongly considered for PFC's. There is little work available on the tritium uptake and retention characteristics of beryllium exposed to a plasma. Work by [illegible] showed.

that retention as indicated by apparent solubility is strongly influenced by the composition and structure of materials, also by its manufacturing history.[4] Further evidence points to trapping that may be induced in damage caused by plasma ions and by neutrons.[5],[6]

Much of the work on PFC materials has been done on thin foils using ion beams, often of greater than prototypical ion energies.[7] A definite lack of detailed information exists on tritium uptake properties of bulk beryllium and other plasma-facing materials. Behavior fundamentally different may be expected in the bulk than in thin foils because of the limited penetration of plasma ions.

Along with uptake and retention characteristics, the release characteristics of PFCs need to be determined. That is, what temperature is needed to cause the tritium to exit the material or structure, what is its chemical form on release, and how fast does it come out? What can be done to augment or suppress release during a thermal transient? This has implications not only for accident consequence estimation, but it may impact the tritium management practices and procedures during operations.

Tritium Permeation to Coolant Streams

When tritium is injected into the surface of a PFC, the immediate tendency is for it to be reemitted from the surface through which it entered. However, because it is unnatural for tritium to exist as single atoms in the gas phase at modest temperatures, the reemission from the surface must be accompanied by recombination of the tritium with either other hydrogen atoms or other atomic complexes susceptible to hydrogen uptake. That is usually a second order process that requires the build-up of a tritium concentration at the surface. Such a surface concentration gives rise to a concentration gradient from the plasma surface to the coolant surface of the PFC structure. That concentration gradient supports diffusion of tritium through to the coolant stream.

Analysts frequently assume that if the coolant is water, any tritium entering the coolant will be eventually lost to the environment because of the difficulty of removing tritium at low concentrations from water. Also, water cooled systems tend to have leaks, and tritiated water can escape, again to the environment. If other coolants are used, the nature of the problem varies, but still there is the threat of leaks to a greater or lesser extent. If the permeating tritium is not lost to the environment, it contributes to the site inventory and generally requires costly systems for removing the tritium from the coolant. Hence, it is of great interest to minimize the permeation of tritium into the coolant streams.

Several features of tritium permeation in PFC materials are of concern from the safety perspective. One is the rate at which tritium will penetrate at steady state, after the system has come to a more or less equilibrium condition. That will normally be a function of the component material, geometry including assembly technique for composite materials, and average temperature. It will also be influenced by the temperature gradient in the material through the Ludwig Soret effect.[8] Normally, that effect will be so small as to be negligible, but for high heat flux components like the divertor it may have a very pronounced effect. Coatings may be applied that may hinder the permeation rate. These may be relatively thick, highly recycling materials on the plasma side that will stop the implanting ions and allow them easy access back to the plasma, thus preventing the base material from seeing any significant particle load. They may also be permeation resistant coatings applied to the coolant side to restrict the flow of tritium into the coolant stream.

Another aspect of tritium permeation that is of concern is the time required to get permeation breakthrough. If it is possible to retard or delay the permeation through a PFC for a time long compared with the operational intervals, then it may be possible to provide a bake-out or other annealing process to drive the tritium from the material before it can appreciably permeate into a coolant stream. Such a delay may be accomplished by deliberately inducing traps, but it is done at the hazard of increased inventory.

One prospect for reducing both inventory and permeation is the creation of a highly recycling material surface. If the surface becomes porous and spongy for a distance only modestly greater than the implantation depth, then atoms or ions implanted from the plasma, when they do diffuse to a surface, exit back to the plasma side, greatly reducing the fraction of implanted atoms diffusing through to the coolant.[5]

Release or Dispersion of Activation Products or Toxins

When materials such as beryllium that have inherent toxicity characteristics are exposed to a plasma, sputtering takes place that may leave the materials in a form that will become hazardous upon opening of the vacuum vessel. For example, beryllium in the solid metallic form used for PFCs is not hazardous. Powdered beryllium, even if it has been oxidized, is respirable and hence must be guarded against in any operations that may allow it to become airborne. Similarly, activation products from impurities or from the materials themselves may be reduced to a state that control is difficult. One of the technical objectives addressed by the experiments using TPE is the evaluation of sputtering characteristics for representative materials. Another is the determination of material mobility after having been sputtered. That is, will it become airborne when the vacuum vessel is opened, or will it undergo chemical reactions that may cause concern.

OTHER ISSUES

Contribution to Waste Streams

It is always advantageous to minimize waste streams. This is particularly important when wastes are hazardous or radioactive. Should wastes become both hazardous and radioactive, then they become hazardous mixed wastes and the problems with respect to safety become compounded. There is not now the ability within the U.S. Department of Energy to legally dispose of hazardous mixed wastes. Fusion designers need to know the potential for generating such wastes in any one of the mentioned categories, caused by the interactions of PFCs with the plasma. The experiment should yield information to help resolve that issue.

Influences of Plasma-Facing Components on Plasma Performance

The potential exists for material from PFCs to enter the plasma, become ionized, and to detrimentally influence plasma performance. One motivation for low-Z number materials for PFCs is the need to prevent synchrotron radiation from electrons not fully stripped from their nuclei. Such radiation can substantially cool the plasma, reducing the energy confinement time. This experiment has the potential to observe changes in plasma emission caused by target material entering the plasma.

Gas puffing, such as now being considered for the ITER divertor design, can be experimentally studied in the dense plasma that TPE can produce. This would provide added data for edge-physics modelers who are just lately able to simulate radiation by impurity gas atoms in multi-dimensional problems. While the configuration of the TPE plasma is markedly different from that in a divertor, measurements made on gas-plasma interactions in TPE could be used to validate analytical models, thus reducing uncertainties in calculations done for ITER or other machines.

TECHNICAL GOALS FOR THE EXPERIMENT

Test Configuration

The sequence of activities now conceived for the project include several technical goals. First will be to ~~conclude~~ decide upon the specific design for the experiment. The first test article will be representative of a leading design for the ITER divertor. That will probably include beryllium as the plasma-facing material with a substrate of copper. Material thickness and joining technique will be prototypic, but because of the limited volume available in the test chamber, other dimensions will be as required to meet chamber limitations.

The target assembly will be actively cooled. For practical reasons, the first coolant used will be water at low pressure. As experience is gained working with the system, other coolants (e.g. pressurized water, pressurized helium, and possibly liquid metals) could be considered. Active cooling is required to sustain the heat loads imposed by the plasma.

The plasmas used will be dominantly deuterium, though tritium will be present when it is useful to do so. Most of the effects studied can be done as successfully with deuterium as with tritium. However, when making permeation measurements and some retention measurements, the detection sensitivity available for tritium will dictate its use. For safety purposes, it may be possible to use only a small fraction of tritium in the plasma for a tracer. Some relatively high tritium fraction plasmas may be used in specific instances. In those cases, it is probable that the full TSTA gas handling system would be used to process TPE effluents and recover the tritium used.

Experiments

The first experiments will evaluate permeation rates and tritium retention under nominal operating conditions. This will be done by operating the plasma column in a temporal pattern and at plasma conditions typical of those expected for the ITER divertor. Permeation rates will be estimated by monitoring tritium concentrations in the coolant stream at regular intervals. Calculations using the TMAP4 code[9,10] suggest that the characteristic breakthrough time for 2 mm of beryllium on a 5-mm thick copper substrate operating at a heat flux of 3 MW/m^2 with a coolant temperature of 373 K (beryllium surface temperature near 1000 K) may vary from about half an hour to two months depending on the amount of trapping that is operative. Steady state permeation rate through a 5-cm diameter interaction area would be about 1.8 Ci/day if the plasma were 1% tritium.

Once the basic operating characteristics of the experiment are evaluated, experiments will be conducted to evaluate tritium retention and release processes for various accident scenarios. Loss of flow accidents (LOFA) or even loss of coolant accidents (LOCA) could be simulated and the evolution of tritium to both to the plasma side and to the coolant side can be monitored. Supplementary heating may be used to simulate gamma-decay heat from neutron activation. Consequence mitigation techniques in design and operating processes can be evaluated.

PROGRAM

This project is expected to last for at least three years. As indicated previously, it will be housed in the TSTA facility at LANL. SNLL personnel will provide initial setup and operating support and contribute to data collection and analysis. Eventually that function will be absorbed by TSTA personnel. Industrial contractor participation will be led by MDC Aerospace as part of their ITER support contract with DOE, administered by SNLL. INEL personnel will contribute to experiment direction, data analysis, and interpretation. Funding for DOE laboratory participants comes directly to that laboratory.

Supporting experiments are expected at INEL and SNLL. INEL operates a smaller scale ion implantation facility with about 0.1% of both the beam area and beam intensity. Its advantage is that the time and cost to conduct small scoping experiments are much less than would be required to conduct such experiments on the TPE. Typical scoping experiments may be to examine permeation and re-emission rates over a range of coating densities for a particular material coating combination. If an optimum is found, that would be selected for evaluation in greater depth in TPE.

It may be appropriate to do dissolution analyses of the plasma facing material for macroscopic tritium profiling over depths of material far greater than would be susceptible to nuclear reaction analysis or other similar techniques. SNLL has a strong history in analyses of this type and other related ones. That laboratory may be called on for support of this kind.

The industrial team led by MDC Aerospace has the responsibility for detailed design and fabrication of the test hardware. This design will be accomplished in consultation with the other participants in response to specific experiment requirements and current ITER design information. Analyses of plasma-material interactions, thermalhydraulics issues, and mechanical responses may be done by any or all of the participants. Results from appropriate analyses will be factored into the design. MDC Aerospace will be supported, in turn, by other partners in the consortium. These include Ebasco Services, Inc., General Atomics, Rockwell International, Westinghouse Electric, and the University of Illinois.

SCHEDULE

This project was initiated late in 1992. It is expected to continue at least through 1996. The general schedule for performing the work is:

Experiment plan and approvals	1993
Fabricate test article and start tests	1994
Report on first test series	1995
Follow on experiments	1996

Specific milestones for technical accomplishments and reporting will be developed as part of the experiment plan and the project management plan.

MOTIVATION AND OUTLOOK

This is an important project that will make important contributions to the body of knowledge needed for successful design and operation of ITER or other tritium burning tokamaks. It deserves and will require a high level of support, both administratively and financially, to achieve success.

Few if any other facilities can simulate the PFC environment as well as TPE can. This simulation includes not only plasma particle flux densities, but particle energies and heat loadings. It also includes use of tritium where needed for isotope-specific effects, observation and sensitivity in measurements. TSTA uniquely provides capability of recovering this tritium for reuse.

Innovative techniques for reducing tritium inventory and optimizing tritium release characteristics need evaluation and verification prior to finalizing the design for ITER PFCs. In many instances these techniques will not be fully submissive to analyses and will require experiments to verify predictions. The TPE, located at TSTA, and supporting experimental systems at INEL and SNLL offer a unique and timely vehicle for accomplishment of these experimental investigations.

Projects like this will integrate industrial partners with DOE laboratory operations. It is the intent of the U.S. DOE, and indeed of ITER, to involve private industries in the development of fusion power. This is in harmony with the need to make emerging technology available to private developers and specifically to involve industry in fusion to pave the way for its commercialization.

It is desirable to maintain the momentum of experience in these kinds of tests. It is time now to take the small-scale experimental work done by INEL, SNLL and other laboratories to the next level of integration and complexity. Single-effect testing needs to be joined by more integrated-effects testing to verify understanding of the mechanisms involved and reveal synergisms that may exist.

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