HIGHLIGHTS OF ITER R&D IN U.S.A.

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ABSTRACT

The U.S. Home Team, in close collaboration with the ITER Joint Central Team and other Home Teams, performs certain technology R&D tasks for the ITER Project. This paper provides an overview and update of the current status of such tasks. The U.S. Home Team devotes most of its effort to the Central Solenoid Model Coil and the Divertor Cassette projects. Additional efforts are devoted to safety issues, limiter components, vacuum vessel welding, ion cyclotron heating components, fueling and tritium components and diagnostics. The main objectives of the Engineering Design Activities phase are expected to be achieved.

1.0 INTRODUCTION

This paper provides an update of a paper (Baker, et al, 1997) that presented U.S. contributions to ITER validating technology R&D. Under the direction of the ITER Director and Joint Central Team, each Home Team performs assigned tasks often in close collaboration with the other Home Teams. The U.S. has co-lead responsibility with the Japan Home Team for the Central Solenoid Model Coil and lead responsibility for the Divertor Cassette project. Efforts in these tasks are highly integrated with all the other Home Teams. The U.S. also plays a major role in safety, ion-cyclotron heating and fueling R&D. Additional U.S. contributions are made in first wall limiter components, vacuum vessel welding, metrology, tritium systems and diagnostics.

2.0 MAGNETICS R&D

2.1 Strand Development

All four parties participate in Nb_3Sn strand qualification and benchmarking of procedures. There has been a major effort to develop manufacturing and efficient QA methods for large scale production, with emphasis on achieving a more uniform and economical product. Production of the thirty tonnes of strand for the model coils was finished and cabled during 1997. The U.S. produced and cabled 4.5 tonnes of internal-tin process, chrome coated strand.

2.2 Model Coil Jacketing

U.S. tasks in this area have provided the 60 tonnes of Incoloy 908 jacket material for the TF and CS Model Coil conductors, as shown in Figure 1. Incoloy 908 is chosen because its coefficient of expansion matches that of Nb_3Sn. However, it is a material that is susceptible to SAGBO (Stress Accelerated Grain Boundary Oxidation.) Therefore the conductor has to be heat treated in a low oxygen environment, and tensile stresses on the exposed jacket surfaces need to be minimized. The low oxygen conditions on the outer surface of the jacket in the U.S. model coil module is provided by heat treating the coil in a 2.9 m dia x 3.8 m tall vacuum furnace maintaining a vacuum of 0.1 mTorr or better. In addition, the conductor outer surfaces is shot peened to assure compressive stresses on the surface. The ID of the conductor, which is mostly in surface compression and where the cable itself acts an effective getter, is thoroughly purged with high purity argon to keep the oxygen and water vapor concentrations below 1 ppm at 1 atm. The specification for these processes were developed as part of the manufacturing R&D program They have been successfully proven in the heat treatment of four layers of the model coil by both the U.S. and Japan teams.

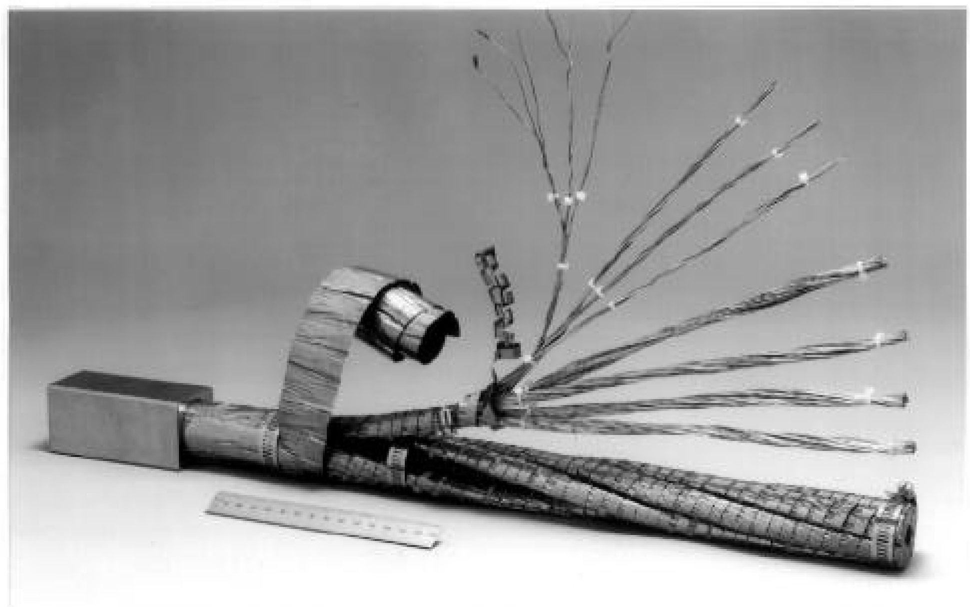

Figure 1 ITER CS Conductor showing cable make-up and Incoloy 908 Conduit

2.3 Joint Development

A sub-scale and full-scale joint development and testing activity has been underway by all four Parties. This is intended to develop manufacturing techniques and demonstrate joint resistance, AC losses, and pressure drop performance. Work in the U.S. has focused on the fabrication of full-scale pre-prototype joints and construction and operation of the PTF (Pulse Test Facility) for test of full scale CICC electrical joints from each Home Team.

The PTF was completed in early 1997 and has been used to test a US pre-prototype lap joint and a Japanese butt-joint. The PTF includes capabilities for sample currents up to 50 kA, magnetic fields up to 6.6 T with ramp rates to +1.5 T/s and -20 T/s, and a cryogenic interface, supplying supercritical helium with flow rates to 20 g/s through each CICC leg at controlled temperatures to up to 10 K and pressures to 10 atmospheres. Parallel and transverse testing of the U.S. pre-prototype joint sample was done at 50 kA transport current at ramp rates to 1 T/s, fields to 4 T and helium flow rates from 3.5-15 g/s/leg. It was found to be completely stable at helium inlet temperatures below 8.5 K. A resistance of 6 nano-ohms was typical of full current and field operation.

2.4 CS Model Coil

The major focus of the U.S. work has been on the inner module of the CS model coil, and on the structure support. The U.S. module is being fabricated by a team headed by the Lockheed-Martin Company in San Diego, CA, with heat-treatment carried out by Wal Colmonoy Company in Dayton, OH, and insulation and terminations, by MIT in Hingham, MA.

The inner module is layer wound with two conductors in-hand, using two grades of heavy-walled square conductor. Each conductor is layer wound and the two conductors are then “corkscrewed” together to make up the two-in-hand winding. The four leads are then bent into position as shown in Figure 2 and layer-to-layer tension plates welded in place. The terminations are then prepared, and the assembly heat treated. Following heat treatment the turns are insulated, and the layer assembled over the previous layers. The 10 layer assembly is ground wrapped and vacuum impregnated before shipment to Japan. As of October, 1997, eight layers have been wound, four layers heat treated, and one layer insulated. Fabrication of the coil will be be completed in July, 1998.

The inner module will be integrated into the cryostat with the outer module and the support structure which applies axial compression to both modules. The coil test will be a joint U.S./Japan responsibility, with participation by the RF and EU home teams.

The test facility, energized with the JT-60U pulsed mode power supplies, has the capability to investigate CSMC performance consistent with the ITER full-scale coil requirements and conductor design criteria. For example, the maximum field at the conductor will be 13 T at 46 kA, and conductor will experience field discharge rates as high as -1.2 T/s for several seconds and field change rates of about 0.3 T/s for relatively long duration.

Figure 2 Inner layer of the inner module of the CS Model Coil

3.0 DIVERTOR R&D

The U.S. is carrying out research to develop the database needed to develop the manufacturing techniques for the plasma facing components and to understand the interaction of the plasma with the divertor surfaces. The U.S. is also the coordinating Party for the development of a full-size prototype divertor cassette and contributing the cassette body and divertor dome. All of our efforts are closely integrated with the efforts of the other Parties and the Joint Central Team. The U.S. team includes Sandia, Argonne, Los Alamos, and Oak Ridge National Laboratories, the University of California at San Diego, the University of Wisconsin, the University of Illinois, Boeing St. Louis, Westinghouse, Boeing-Rockwell, Brush-Wellman, Surmet, Plasma Processes, and Karta Technologies.

3.1 Research and Development Activities

The U.S. efforts have focused on the development of reduced activation techniques for joining beryllium and tungsten to copper alloys, development of new heat sink designs that improve the heat removal capability, and plasma wall interaction studies including erosion, disruptions, and tritium retention. Other efforts have included development of plasma spray as a method for application of plasma facing materials or in-situ repair of such materials and construction of medium size mockups of plasma facing components for the divertor.

Since beryllium forms intermetallics with most other materials, we have focused our joining efforts on the development of diffusion barrier coatings and the use of the few elements that do not form intermetallics with beryllium. We have studied the use of aluminum, and aluminum alloys as joining materials. The aluminum bonding has been performed at temperatures from 450 to 650 C with 100 MPa pressure. The bonding temperature must be kept low to prevent reduction of the strength of the copper alloy heat sink. We have just completed fabricating the second round of specimens for high heat flux testing. These samples include the use of aluminum-beryllium alloy interface layers since the pure aluminum layers were limited by the strength of aluminum. The most successful joints have survived up to 3000 thermal cycles at 10 MW/m2. Both the Al Be Met and Al interlayers performed equally well. Short pulses (10s) of 20MW/m^2 were applied to simulate disruptions. The surface melted but there were no interface failures.

The hypervapotron has been shown to have the highest critical heat flux and the lowest pressure drop in testing conducted by the European Home Team and the U.S. Home Team. The U.S. is fabricating a hypervapotron with the internal fins attached to the sidewall of the channel. The resulting reinforcement of the top surface allows it to be thinned to 1 to 1.5 mm. Another variation being investigated jointly with the Russian Home Team is the use of porous coatings inside the water cooling channel. Preliminary test results indicated the porous coatings could enhance the heat removal without significant pressure drop over a smooth wall design.

One of the advantages of using metallic materials for plasma facing surfaces is the possibility of using plasma spray to repair or rebuild eroded or damaged surfaces. Los Alamos National Laboratory has developed a technique for low pressure plasma spray of beryllium that produces coatings that are nearly fully dense (>98%) and have excellent thermal conductivity (>98% of theoretical at room temperature). Mock-ups of the ITER first wall have been coated with 10 mm of beryllium using plasma spray. Thermal testing of those samples revealed no damage after up to 3000 cycles at 1 MW/m^2 (nearly four times the ITER requirements).

The PISCES device at the University of California, San Diego, has been placed in a clean room to allow it to be used to study beryllium erosion under conditions typical of ITER. It was found that the erosion of beryllium was greatly reduced (<3%) when there was carbon on the surface of beryllium. This effect was particularly prominent at elevated temperatures (>250 C where the ITER divertor operates) but very small at low temperatures. Further studies of erosion on other mixed materials (e.g., C on W, W on Be, etc.) are being conducted. The effect of mixed materials on hydrogen isotope trapping and release are also being studied. The DiMES probe at DIII-D was used to measure erosion in the divertor region. The results of the PISCES and DiMES studies were used to benchmark the REDEP code developed at Argonne. This code has been used to predict erosion of plasma facing materials in ITER.

Argonne has used the results of disruption simulations in the RF and disruption studies on DiMES to benchmark the A*THERMAL code. This code includes influence of material erosion on the incident plasma during a disruption. The incident power is reduced by radiation from the eroded material in the plasma. For metallic materials another code (SPLASH) has been developed to predict the loss of melted material. This code includes MHD effects, currents in the melted material, and the effect of the plasma pressure on the surface.

The Tritium Plasma Experiment (TPE) device operated by Sandia at Los Alamos has been used to study tritium retention, release and permeation in beryllium. The heat and particle fluxes were typical of those in ITER. Exposures of up to 40 hours were performed. The results show that tritium retention is very low in beryllium. Codeposition of beryllium and tritium does occur but the retention is very low. Retention of tritium in irradiated carbon fiber reinforced graphite has also been studied.

In collaboration with the Russian Federation, an irradiation program in the SM-2 reactor is providing the irradiation performance data needed to select the appropriate copper alloy for various high heat flux components. A unique feature of the materials-engineering for ITER is the application of relatively thick sections of laminated materials. The limiter and first wall/shield structure is bonded to a copper alloy heat sink; the plasma-facing material is, in turn, bonded to the heat sink to provide a triplex laminated structure. A variety of methods of bonding plasma facing materials (Be, W, C/C composites) to the copper alloy heat sink are being investigated. Although the outer bonded layers are not load bearing, crack-initiation and growth in these layers and/or partial delamination in the bonded region could seriously impact the integrity of the underlying structure. A collaborative program with the RF is in progress to compare the irradiation performance of bonded materials prepared by various fabrication routes and to develop a design database for performance analysis of these unique materials.

3.2 Full-Size Divertor Prototype Development

One of the main objectives of the ITER Engineering Design Activities is to produce a full-size prototype of a divertor cassette. This task is shared among all the parties. The responsibility of each party is shown in Figure 3. The U.S. has the responsibility to coordinate the design, fabrication, assembly, and testing of the prototype. The coordination of the designs among the Parties is being handled through the use of interface control drawings. These drawings show the details of the connections between the cassette body and all of the plasma facing components and the clearances between the components. The drawings are being maintained by Boeing, St. Louis.

The cassette body will be fabricated using cast and HIPed stainless steel. The removal of the nuclear heating in the body is accomplished using channels bored through the thin (toroidal) direction of the body. These cross channels are connected by surface channels that also carry the coolant to the plasma facing components. Two of four castings have been made. The cast material has about 80% of the strength of wrought material. Contracts are in place with industry for the fabrication of the full-size prototype.

4.0 SAFETY R&D

Most of the work is performed by the U.S. Fusion Safety Program (FSP) at the Idaho National Engineering and Environmental Laboratory with smaller contributions from University of Wisconsin (UW) and North Carolina State University (NCSU).

The U.S. has tested several fusion relevant materials for their chemical reactivity in air and steam. Work during the ITER EDA has been focused on testing W and various forms of Be (e.g., dense, irradiated, and plasma sprayed) using improved instrumentation and testing techniques. The temperature-dependent hydrogen generation rates of fusion relevant materials are used to determine whether the hydrogen generation during specific accident scenarios is below the ITER limit of 10 kg.

The U.S. plays a major role in defining the activation product source term – radioactivity release in terms of mass, particle size, and chemical form – for ITER accident analysis. Current tasks involve characterizing tokamak dust from DIII-D and TFTR. Work is also underway at NCSU to characterize the mass and particle size distribution of 316 SS, C, Cu alloy, and W as a result of ablation during a disruption.

The U.S. also uses experiments and analytic models to determine the tritium inventory in ITER in-vessel components. Our code, TMAP4,has been used to interpret the tritium inventory saturation behavior seen in Be experiments in the U.S. and Europe so that accurate estimates of tritium inventory in ITER can be developed. FSP experiments address tritium inventory in ITER relevant materials and mobilization behavior in accidents. Additional experiments examine the influence of surface contaminants on inventories in W and Be.

Accurate safety analysis of ITER accidents requires integrated systems-level transient state of the art computer codes that can model all the important phenomena from accident initiation to radioactive releases to the environment. The U.S. is focused on developing three safety computer codes: ATHENA, CHEMCON, and MELCOR. MELCOR, developed by Sandia National Laboratories (SNL), is a code that models thermal hydraulic and source term behavior during severe accident conditions. Upgrades have been made to MELCOR during the ITER EDA to account for chemical reactivity of ITER relevant materials, water and air condensation and freezing under cryogenic conditions, ice formation on cold structures, tritiated water (HTO) transport, aerosol turbulent deposition, thermal radiation in gas enclosures, and liquid metal/water interaction models (at UW). All three codes were used in the recent ITER Non-site Specific Safety Report (NSSR-2).

5.0 ION CYCLOTRON HEATING R&D

R&D for the ITER ion cyclotron (IC) heating system has been focused on the optimization of the ion cyclotron antenna configuration. Detailed studies have been carried out to optimize critical antenna dimensions (e.g., current strap width, Faraday shield thickness, septum length and thickness) subject the constraint that the IC launcher must fit in through a main horizontal port. Optimization studies resulted in about a 10 - 15% decrease in the peak voltage. This has put the voltage (and electric field) levels into the range of operation of some present-day tokamaks. In order to verify the voltage-handling capabilities of the ITER antenna, an R&D prototype is being built by the U.S. and Japan Home Teams that will be tested in vacuum to determine the peak rf voltage it can sustain. Longer-range plans involve tests of the antenna with advanced tuning mechanisms from EU, and possibly tests with plasma.

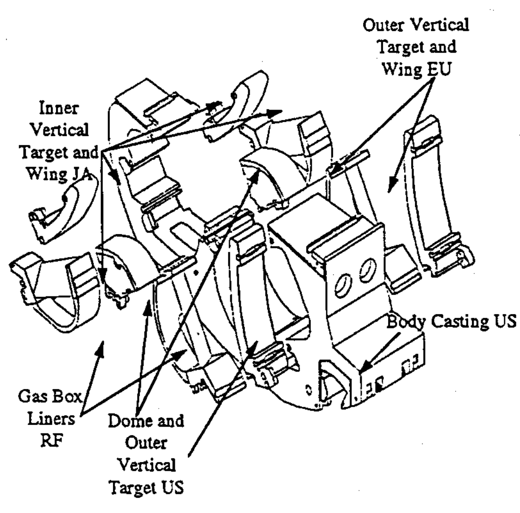

Figure 3 Divertor Distribution of Prototype Fabrication Among the Four Parties

6.0 FUELING R&D

The U. S. is responsible for most of the R&D and design activities for the ITER plasma fueling system, which consists of a gas injection system (GIS) and a pellet injection system (PIS). A full-scale prototype of the GIS has been set up at ORNL and experiments have quantified system throughputs and response times for hydrogenic and impurity gases. Initial results indicate the specified response time of 1 s for DT gas fueling can be achieved but modifications are needed to the GIS system (moving the gas injection valve much closer to the vacuum vessel) to achieve the desired 100 ms response time for impurity gases injected at low flow rates.

During the EDA there has been much progress in the development of pellet fueling components for ITER. The first ever extrusion of solid tritium and the subsequent production of the world's largest cryogenic pellet has been accomplished; this work was done under an ORNL/LANL collaboration. The ~ 8 mm tritium pellet would result in about a 5-10 % density perturbation to ITER. ORNL has also demonstrated the extrusion of solid deuterium at mass flow rates of 0.26 g/s which are in excess of the ITER requirements. The Russian Federation has operated a screw-type extruder for about 1 hour, continuously extruding 2 mm solid hydrogen. Also developed during the EDA is an innovative fueling concept called isotopic fueling, where tritium-rich pellets fuel the plasma interior and deuterium gas is used to fuel the edge region. This concept can substantially reduce tritium throughput in the torus and in-vessel tritium inventories. Finally, ORNL is conducting R&D on pellet transport in curved guidetubes in support of recent positive results from pellet injection into the high field side of ASDEX-U and vertical injection of pellets on DIII-D. Initial results with 2.7 mm and 10 mm deuterium pellets indicate substantial velocity limits for intact pellets as they negotiate curved guidetubes with radii of curvature in the range 5-100 cm.

7.0 FIRST WALL/SHIELD R&D

The U.S. R&D effort in support of the first wall-shield-blanket has focused on the development and testing of the first wall for the limiter. The limiter first wall must accommodate a maximum power of 5 MW/m2 (10 MW/m^2 peak) during start-up and shut-down compared with 0.5 MW/m^2 for the rest of the first wall. The participating institutions are Argonne National laboratory, Boeing, the University of California - Los Angeles, the University of Illinois, the University of Wisconsin, and Westinghouse Electric.

The U.S. effort has addressed fabrication of Cu/SS plates using candidate copper alloys, testing of the properties of the Cu/SS joints, and fabrication of first wall mock-ups which include thin walled coolant channels in the copper alloys. The joints were fabricated from Glidcop" CuAl25 or CuNiBe HP3 joined to 316L SS by Hot Isostatically Pressing (HIP). Following fabrication, the plates were examined by nondestructive examination techniques and tested for mechanical performance using a variety of testing techniques. The results indicate that the HIP bonding technique is a reasonable choice for plate fabrication for ITER first wall application.

8.0 VACUUM VESSEL R&D

The vessel is divided toroidally into sectors for installation with the associated TF coil. Field welds are required in both shells to join adjacent sectors. Demonstration of remote welding and cutting techniques for these field joints is critical for both the construction and remote maintenance of ITER. To this end, the goals of one of the seven major EDA deliverables are the fabrication of a full scale prototype vessel sector and demonstration of remote cutting, welding, and inspection of the vessel field assembly joint. The U.S. has the lead for accomplishing the second goal and will provide a demonstration of the remote cutting and welding system processes on the full scale, partial joint mockup. R&D studies by Rocketdyne Division of Boeing have led to the selection of the Narrow Gap-Tungsten Inert Gas (NG-TIG) process for welding and the plasma arc process for cutting. The NG-TIG process makes reliable, X-ray quality welds in the 40 mm thick vacuum vessel material with a weld joint gap width of only 5 mm. Figure 4 illustrates a computer simulation of the robotic welding system operating on the full scale partial field joint mockup. The robot consists of a five axis manipulator integrated with a track mounted vehicle. A single monorail track oriented on the plasma side of the field joint can be used to weld both the plasma side and coil side facesheets.

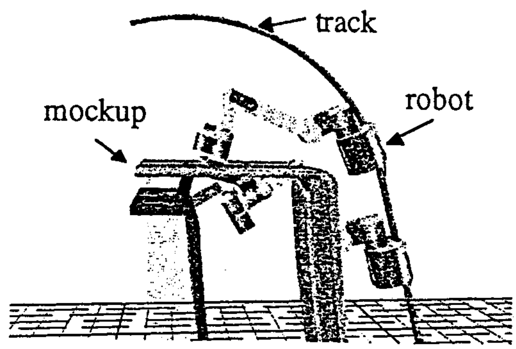

Figure 4 Computer simulation showing robotic welding system for vacuum vessel mockup

9.0 IN-VESSEL VIEWING/METROLOGY R&D

A metrology system, capable of achieving sub-millimeter accuracy, is under development at the Oak Ridge National Laboratory (ORNL). It must operate in the torus environment under high gamma radiation (10^4 Gy/h), high vacuum (10^5 torr), elevated temperature (200°C), and a magnetic field of 6.2 T. Hence, the system must be remotely deployed. A coherent, frequency modulated laser radar system, and a remotely operated deployment mast are being developed to meet these requirements. The metrology/viewing sensor consists of a compact laser transceiver optics module which is linked through fiber optics to the laser source and imaging units, which are located outside the torus. The deployment mechanism is a telescopic-mast positioning system.

The sensor will be capable of range measurements up to 20 meters with a range precision of 10 micrometers. Gamma radiation exposures up to 10^7 Gy were conducted on a crystal component of the sensor and on the polarization maintaining fiber, at ORNL. No significant impact to the data transmission characteristics of these components was observed. Additional radiation testing is planned for the acousto-optical module, along with high field magnet tests.

Finally, the capability of the system to produce visual images is also being investigated. A metrology map of surface features can be converted into “picture quality” images for visual assessments of surface patterns. In this manner, metrology data obtained without the use of an illumination source can be used to produce high-quality surface images.

10.0 DIAGNOSTICS R&D

Assuring understanding and documentation of the operation of the ITER plasmas demands diagnostics with at least the same quality of temporal and spatial resolutions as for present devices, and this must be achieved with complex access to the plasma and substantial radiation levels. It is expected that because of the very long pulses, many plasma measurements will be incoporated into the control network so that the measurement capability requires a high degree of reliability. Among U.S. responsibilities in design of diagnostics for ITER are the Thomson scattering system for the electron temperature and density in the edge; the set of active spectroscopy instruments, i.e. those dependent on a neutral beam to provide source particles for measurements of the spatial distributions of key parameters such as ion temperature, rotation, current density distribution and hydrogen-isotopic ratio in the core; and the neutron flux and fluence measuring systems. Component testing for magnetic diagnostics and innovation and testing of prototype diagnostics are also under way to find new ways of measuring the confined alpha-particles.

11.0 TRITIUM PLANT R&D

The U.S. contributes to ITER Tritium Plant R&D in a number of areas. These include torus exhaust fuel cleanup (permeators and impurities processing), vacuum pumping, isotope separation, tritium storage, analytical techniques, breeding blanket processing, tritium handling operational techniques, and fuel cycle computer modeling. The most recent advances have been in two areas: tritium storage and impurities processing.

Performing an inventory of a hydride storage bed by driving the hydrogen isotopes off of the bed can be time consuming. However, an inventory can be performed based on the decay heat of tritium. A self-assaying bed has been constructed which acts as a calorimeter so that the amount of tritium on the bed can be determined at any time by measuring the bed's temperature.

One of the most daunting problems facing the tritium plant is the recovery of tritium from expected torus exhaust "impurities" such as methane and water. For this purpose a palladium membrane reactor (PMR) has been designed and constructed. A catalyst is used to promote reactions such as methane-steam reforming ($CH_4 + H_2O \rightarrow 3H_2 + CO$) and the integrated permeator removes the hydrogen isotopes as they are generated. This has proven effective for recovering hydrogen isotopes from streams containing various concentrations of water and methane. Over many months of operation it has been found to be reliable and relatively easy to operate.

Decontamination factors (tritium recovered/tritium fed) as high as 108 have been measured.

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This report is an account of work undertaken within the framework of the ITER EDA Agreement. Neither the ITER Director, the Parties to the ITER Agreement, the U.S. DOE, the U.S. Home Team Leader, the U.S. Home Team, the IAEA or any agency thereof, or any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the parties to the ITER EDA Agreement, the IAEA or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the ITER Director, the Parties to the ITER Agreement, the U.S. DOE, the U.S. Home Team Leader, the U.S. Home Team, the IAEA or any agency thereof.

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REFERENCES

Baker, Charles C., and the US Home Team "Status of ITER R&D in the U.S.A.", accepted for publication in *Fusion Engineering and Design*, special issue,