

Tritium Handling Requirements and Development for Fusion

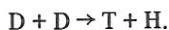
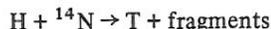
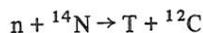
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Invited Paper

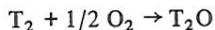
Abstract—Tritium technology development for the U.S. Department of Energy fusion program is taking place principally at three laboratories: Mound Facility, Argonne National Laboratory, and Los Alamos National Laboratory. This paper reviews major aspects of these three programs and discusses aspects of the tritium technology development at other laboratories in the U.S. Facilities and experiments discussed include the Tritium Effluent Control Laboratory and the Tritium Storage and Delivery System for the Tokamak Fusion Test Reactor at Mound Facility, the Lithium Processing Test Loop and the solid breeder blanket studies at Argonne National Laboratory, and the Tritium Systems Test Assembly at Los Alamos National Laboratory. Work at the Lawrence Livermore National Laboratory and the Oak Ridge National Laboratory will also be discussed.

I. INTRODUCTION

THREE AREAS of fusion technology deal with tritium handling and containment. These are the main fuel cycle including reprocessing of the reactor exhaust gas, tritium breeding and extraction from blanket systems, and tritium safety. Each area is being investigated under the sponsorship of the Office of Fusion Energy. We anticipate that early-generation fusion reactors will use the deuterium-tritium (DT) fuel cycle. The tritium-fuel component is a hydrogen isotope of mass three. This material is radioactive, decaying by emission of a beta particle of 5.6-keV average energy and a half-life of 12.3 years. Tritium chemically behaves like ordinary hydrogen; reacts with oxygen to form water, with nitrogen to form ammonia and it will exchange with the hydrogen in hydrocarbons to form tritiated hydrocarbons. Because of its relatively short half-life, tritium is not an abundant isotope. Tritium is produced in nature by cosmic-ray bombardment of the earth's upper atmosphere. Neutrons, protons, and deuterons cause tritium producing reactions such as



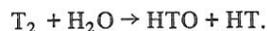
Tritiated water is formed in the atmosphere by tritium oxidation or tritium exchange with hydrogen in water through a variety of complex reactions that can be summarized as



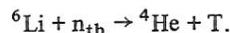
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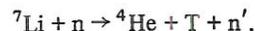
and



Tritium normally is produced in nuclear reactors by the thermal neutron irradiation of ${}^6\text{Li}$, where the following reaction occurs:



Tritium also is produced by the fast neutron



These reactions will be used to breed tritium in a fusion reactor blanket. This blanket might be a liquid-lithium-metal blanket with an adjusted ${}^6\text{Li} : {}^7\text{Li}$ ratio, or it might be a solid-lithium material such as lithium oxide, a lithium-lead, or a lithium-aluminum alloy where the thermonuclear neutrons would be captured by the lithium, thus breeding tritium.

The primary fusion reaction in a DT plasma is the $\text{T}(\text{D},n){}^4\text{He}$ reaction with an energy release of approximately 17.5 MeV, a neutron kinetic energy of 14 MeV, and a ${}^4\text{He}$ energy of 3.5 MeV. To obtain useful power from the energetic neutron, the reactor designers must provide a surrounding blanket in which the neutron energy is moderated by multiple collisions with the blanket materials and eventually is captured by one of the blanket or shielding components. In addition, new tritium fuel must be produced by a nuclear reaction to supply fuel to the plasma. Both objectives are accomplished by the use of lithium in the blanket. Because natural lithium is composed of two isotopes (7.42-percent ${}^6\text{Li}$ and 92.58-percent ${}^7\text{Li}$), two nuclear reactions occur. At thermal neutron energies, the reaction ${}^6\text{Li}(n,\alpha){}^3\text{H}$ predominates with an exothermic energy release of 4.79 MeV. At higher neutron energies, the reaction ${}^7\text{Li}(n,n'\alpha){}^3\text{H}$ increases in importance but has an endothermic Q -value of -2.47 MeV. The thermal energy from the nuclear reactions in the blanket and from the alpha-particle heating of the plasma eventually will be converted to electrical power with an efficiency of ~30 percent; therefore, large quantities (~1 kg/day) must be supplied to the reactor. The quantity of fuel that must be supplied to the plasma is much larger than the amount consumed by nuclear fusion, because only a fraction of the fuel reacts during its residence time in the plasma. Based on present forecasts of particle confinement time, design studies on magnetically confined reactors project from <1 percent to 10 percent "burn-fraction" for the fuel. Therefore, it is necessary to provide rapid fuel reprocessing and re-

circulation of the unburned fuel for reinjection into the plasma.

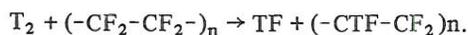
Fuel recirculation requires a series of chemical and physical processing steps including evacuation of the plasma chamber, removal of impurities, adjustment of the DT ratio, removal of protium (^1H) by a hydrogen-isotope separation technique, and storage of the separated fuel components before reinjection of the fuel to the plasma.

All fusion reactor designs propose the generation of new tritium fuel by the nuclear reaction of the neutrons originating in the plasma with the lithium-bearing materials in the blanket. Suggested lithium-bearing materials are liquid metal, metal alloys, fused salts, and ceramic compounds. Each material retains some equilibrium tritium level, below which tritium cannot be extracted. This equilibrium blanket level contributes significantly to the total tritium inventory in a fusion plant.

Liquid lithium has been proposed as both the breeder material and the heat transfer fluid; however, it does have a high affinity for tritium. It also poses several safety problems because of the highly reactive chemical nature of lithium. Lithium ceramic compounds have been proposed as breeder materials; here the tritium solubility in the breeder material is fairly low at the high operating temperatures of the blanket. It is proposed that this tritium would continuously diffuse from the ceramic and be removed in the stream of an inert gas. Tritium diffusion, either as the chemical species T_2 or as T_2O , through the ceramic is relatively slow; consequently, the particle size must be kept small to prevent excessive tritium "hold-up."

A DT burning fusion machine will require large tritium inventories. The Tokamak Fusion Test Reactor (TFTR) at the Princeton Plasma Physics Laboratory will have a tritium inventory of 5 g or less, and there will be no on-site fuel reprocessing. However, the tritium inventory at a next generation machine (perhaps the Fusion Engineering Device (FED), the International Tokamak Reactor (INTOR), or the Engineering Test Facility (ETF)) will require a few kilograms. This inventory, in itself, will be a large source of radioactivity (1-g tritium $\approx 10^4$ Ci) at a fusion site. Adequate personnel and environmental protection and safety systems must be integral parts of the tritium facility at such a fusion machine. The development and evaluation of these safety systems must be accomplished in a timely manner so that these issues do not pose a threat to public acceptance of fusion as an energy source.

Other special problems introduced by tritium are related to radiation damage and chemical compatibility of materials used in tritium facilities. The tritium itself is a radiation source, although the tritium beta particle is a low energy source. This beta particle can cause radiation degradation in some materials, notably in the hydrocarbons, including elastomers that might be used for gaskets, valve seats, etc. A special problem exists with the use of perfluorohydrocarbons, such as Teflon and viton, because the tritium undergoes exchange reactions with the fluorine



This reaction leaves an internal radiation source as a component of the elastomer; however, a more serious problem is generation of the extremely corrosive and radioactive tritium fluoride (TF). For this reason, the use of fluorinated hydrocarbons must be avoided in the design of tritium systems. Also, because tritium will cause hydrogen embrittlement, as do ordinary hydrogen and deuterium, materials used in a tritium facility must always be those which are known to be resistant to hydrogen embrittlement.

II. CURRENT ACTIVITIES IN TRITIUM TECHNOLOGY

A. Los Alamos National Laboratory

In 1977 the Office of Fusion Energy, U.S. Department of Energy, funded the Los Alamos National Laboratory to design, construct, and operate the Tritium Systems Test Assembly (TSTA) [1]. The principal objectives of the TSTA Project are to

- 1) demonstrate the fuel cycle for fusion power systems;
- 2) develop, test, and qualify equipment for tritium service in the fusion program;
- 3) develop and evaluate personnel and environmental protection systems;
- 4) provide a facility that will yield a reliable data base for tritium handling systems for future fusion facilities;
- 5) demonstrate long-term safe handling of tritium with no major releases or incidents;
- 6) investigate and evaluate the response of the fuel cycle and environmental packages to normal, off-normal, and emergency situations; and
- 7) develop tritium-compatible components with long-term reliability.

The TSTA schedule calls for the facility to be operational by the end of April 1982. Modifications to an existing building at Los Alamos have been completed (Fig. 1), and installation of equipment for specific subsystems now is under way. The project is more than halfway through the design-construction phase and currently is on schedule for the 1982 operational milestone. The goal of the TSTA project is to provide an extensive data base for the designers of the first large-scale DT burning fusion machine, probably the FED or the INTOR. The FED concept is being developed through the Engineering Test Facility Design Center (TFDC) at Oak Ridge National Laboratory. Optimistic estimates indicate that a FED could be operational during 1991-1992 but would require a detailed design phase during 1983-1987. To make a significant contribution to the FED data base, TSTA must become operational by early 1982. The INTOR project is a joint international effort involving the United States, Japan, the USSR, and the European Community, and its goals and objectives are quite similar to those of FED. At this time, the four INTOR participants are engaged in workshop and conceptual design meetings. Realistically, we cannot predict that INTOR can be built any faster than FED can. Therefore, the current TSTA schedule will also provide the necessary data base for INTOR.

The TSTA will consist of a large interactive gas loop (Fig. 2) that simulates the proposed fuel cycle for a fusion facility. The reactor torus will be simulated by a vacuum vessel into which gas mixtures are introduced at the compositions and pressures predicted for an actual reactor torus at the end of a burn cycle. This gas mixture, primarily $(\text{D},\text{T})_2$ containing a variety of impurities, must be evacuated through a prototypical vacuum system; impurities must be removed from the $(\text{D},\text{T})_2$; and isotopic separation must be performed to produce D_2 , T_2 and DT. The resulting gases then will become the fuel components that are injected into the reactor (vacuum vessel) for the next burn cycle. The gas loop is designed to handle up to 360 g-mol/day (1800 g) of DT. This flow will provide cycle-operating experience on a scale similar to that being considered for FED and INTOR. In addition to the gas loop, all safety and experimental systems associated with such an extensive tritium facility are under consideration. To accomplish the



Fig. 1. The TSTA building.

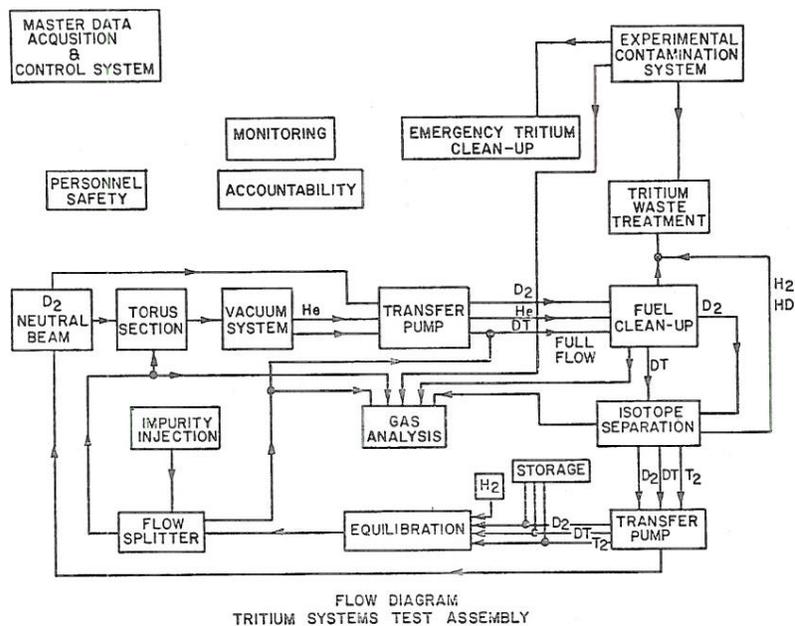


Fig. 2. The TSTA process loop showing subsystem interactions.

program goals, an on-site tritium inventory of approximately 150 g is required. This report discusses the current status and design of each major TSTA subsystem.

1) *Vacuum Facility (VAC)*: This system consists of a large vacuum vessel with associated duct that leads to the vacuum

pumps being evaluated for use in the fusion program. Three cryo-based vacuum pumps, candidates for the primary vacuum pumps on a fusion reactor, are being evaluated at TSTA. These are "compound" pumps because they are two stage. The first stage is cryocondensation of hydrogen on a metal surface cooled

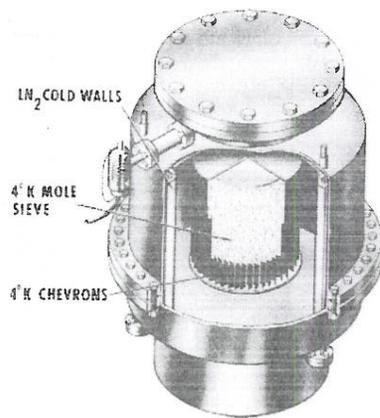


Fig. 3. The Los Alamos TSTA compound cryopump.

to or near liquid-helium temperatures. This stage will pump all the hydrogen isotopes, but not the helium isotopes. In the first pump (Fig. 3) built by the Los Alamos National Laboratory [2], the helium is pumped by cryosorption on a molecular sieve surface cooled to liquid-helium temperatures. The second pump, developed at Brookhaven National Laboratory [3], will pump helium by cryosorption on a charcoal surface cooled to liquid helium temperatures. The third pump, built by the Lawrence Livermore National Laboratory, will pump helium by argon cryotrapping. A pump very similar to that being built for TSTA by Lawrence Livermore National Laboratory has been described by Batzer *et al.* [4]. In this pump a fine spray of argon gas will be injected into the pumping area and will condense as a solid on a helium-cooled metal surface. The helium gas will be trapped and thus pumped by the argon ice formed on the cold surface. All three pumps have been built and are undergoing performance testing. The Los Alamos National Laboratory built pump has demonstrated that a compound pump can simultaneously pump helium and hydrogen isotopes. The separation of hydrogen, on the condensation panel from helium on the cryosorption panel, which is extremely sharp, can be maintained by careful temperature and pressure control during regeneration. The helium panel is first regenerated, then the cryocondensation panel can be warmed to remove hydrogen from the pump. Complete evaluation of all three pumps will continue.

The TSTA vacuum facility also will include a complete regeneration system to be used during regeneration of the cryopump. The regeneration system has been described by Coffin and Walthers [2].

2) *Fuel Cleanup (FCU)*: The primary function of the FCU is to separate all of the other atoms present from the hydrogen isotopes in the gas stream from the VAC facility. This step is essential. If other molecular species, such as tritiated water, methane, ammonia, and argon, were present in the hydrogen-isotope stream, they would freeze and plug the low temperature distillation columns of the Isotope Separation System. The FCU not only must separate $(D,T)_2$ and $(H,D,T)_2$ from the reactor of offgas contaminants, it also must recover as $(H,D,T)_2$ all hydrogen isotopes that are chemically combined with other atoms in the reactor offgas, i.e., $C(H,D,T)_4$, $N(H,D,T)_3$ and $(H,D,T)_2O$. Table I shows the predicted feed stream flows to the FCU. The hydrogen molecules can be separated from the other molecular species by adsorbing the impurities at 75 K on

TABLE I
IMPURITY GAS LOAD FOR TSTA
(based on 360-g moles/day DT flow)

Element	Mol %	Species	Grams/day
He	2-20	He	29-290
H	1	HD, HT, H ₂	7
C	0.02-0.01	C(D,T) ₄ , C ₂ (D,T) ₂ , CO ₂	1.7-9
N	0.01	N(D,T) ₃ , N ₂ , CN, NO _x	10
O	0.5	(D,T) ₂ O, CO ₂ , O ₂	57
Ar	0.00006-0.05	Ar	0.17-14

a 5-A molecular sieve. (The tritiated ammonia and water are removed from the offgas before it enters the molecular sieve.) Regenerating the molecular sieve requires additional processing. These considerations have led to two proposed processing schemes (Fig. 4). In one system, a front-end hot-metal bed (uranium at 1170 K) removes carbon, nitrogen, and oxygen from the feed stream by converting them to uranium carbides, nitrides, and oxides and releasing the associated hydrogen isotopes as gas. Periodically, the uranium will become saturated with impurities and must be replaced. The second front-end system has a catalytic reactor to convert any free oxygen in the feed stream to $(D,T)_2O$, which is then removed, along with the ammonias and carbon dioxide by freezing. These two front-end packages would be operated alternately. Each front-end system is backed by a cryogenic package where argon and any other trace impurities are removed by adsorption on molecular sieve, thus producing a pure stream of hydrogen isotopes to feed the isotope separation system. The FCU has been described in detail in a previous paper [5].

3) *Isotope Separation System (ISS)*: At TSTA, cryogenic fractional distillation [6], [7] is being used for hydrogen isotope separation. A system that uses four interlinked columns, with chemical equilibrators located between columns 1 and 2 and between columns 2 and 4 (Fig. 5) has been designed. The system is sized to handle the full flow appropriate to FED or INTOR, i.e., 360 mol DT/day. It also will handle the simulated flow from the neutral beam channel vacuum pumps (~ 275 mol D₂/day). Fig. 5 shows the flow paths and purities of the major components in each of the four output streams. Refrigeration will be supplied by a central cryogenic refrigerator that provides helium gas at 20 K. In case of refrigeration loss, the ISS contains two systems for storing gas evolved from vaporization of the hydrogen liquids.

First, a large surge tank allows the total contents of the ISS to be stored, as a gas, at a pressure <100 psi. Second, each column has access to a vessel filled with uranium powder that can hydride the gaseous hydrogen isotopes and thus store them as solids. The contents of each column can be stored individually during periods of planned shutdown. The ISS is totally under double containment. The distillation system now is installed at TSTA. Performance tests by the manufacturers before shipment of the columns to the laboratory demonstrated an excellent separation of H₂, HD, and D₂ in the columns. Tests with tritium will not, of course, be performed until tritium operation of TSTA occurs.

4) *Transfer Pumps (TPU)*: The transfer pumps at TSTA will provide circulation and transport for mixtures of hydrogen and helium isotopes from one portion of the flow loop to another. An early decision was that all process-wetted compo-

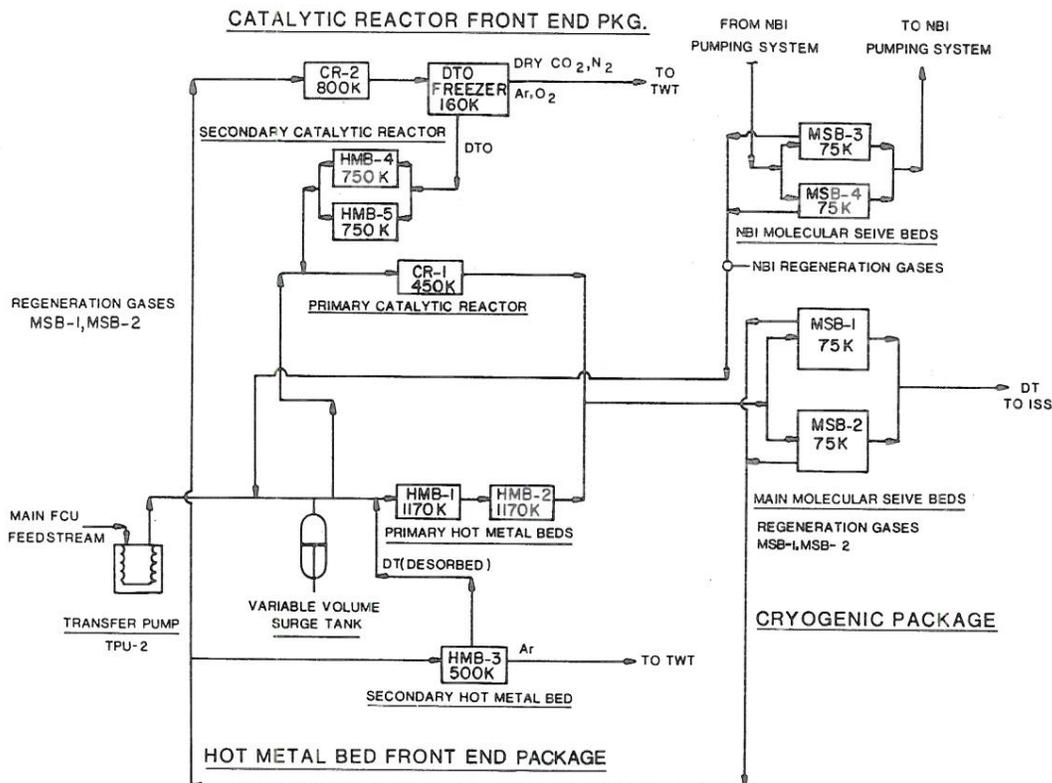


Fig. 4. TSTA fuel cleanup flow schematic.

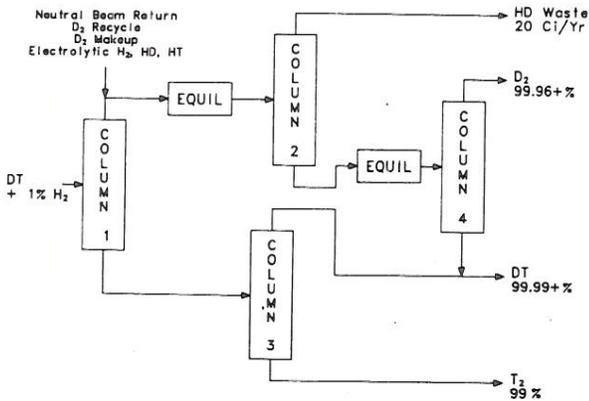


Fig. 5. Flow diagram for the ISS distillation columns. The ISS is designed to provide pure D₂ for neutral beams, HD waste free of T₂, as well as streams for reactor refueling and materials studies.

nents of transfer pumps would be of metal or inert carbon construction. Mechanical motion of internal parts is transmitted by flexible metal membranes (diaphragm or bellows), by magnetic transmission (canned motors), or by magnetically confined sealants (Ferrofluids). Most TSTA gas transfer requirements are met by a metal-bellows pump (Model MB-601, manufactured by Metal Bellows Corporation, Sharon, MA) that incorporates a replaceable all-metal check valve assembly designed and tested at the Los Alamos National Laboratory. The basic pump has two heads, which can be plumbed to operate in series, in parallel, or independently. Fig. 6 shows our concept of the TSTA metal-bellows transfer pump housed in the secondary containment glove box. Other pumps being considered include an all-metal, bellows-sealed stainless steel scroll pump

and a canned-motor ferrofluid-sealed roots blower. These pumps are being tested and installed at TSTA.

5) *Emergency Tritium Cleanup (ETC)*. This system will process all of the air in TSTA if a gaseous tritium release to the facility occurs [8]. It will recover most of the released tritium, thus reducing losses and environmental impact. The TSTA cell contains 3000 m³ of building atmosphere that would become contaminated with tritium if an accident caused the primary and secondary containments to be breached. The flow rate through the ETC is 0.65 m³/s. The ETC will be an automatically actuated, room-air detritiation system based on a precious metal catalytic recombiner where hydrogen isotopes are oxidized to water. The water is collected, partly as liquid water and by adsorption on molecular sieve beds. The flow path for gas through the ETC is shown in Fig. 7. A 100-g T₂ spill into the facility would give an initial concentration of ~355 Ci/m³ in the cell, and the ETC would reduce the room level to 40 × 10⁻⁶ Ci/m³ within 24 h. The ETC at TSTA was designed with individually purchased components, and the assembly and installation of these components, including instrumentation, is being done by the Los Alamos National Laboratory. The ETC equipment has been bought, and the installation of this system is nearing completion.

6) *Tritium Waste Treatment (TWT)*. This system provides routine processing of all gaseous effluents generated at TSTA to remove tritium from these effluents before they are released to the environment. It is based on the design of a similar system that has been operating for 5 yr [9] and that has been discussed recently by Nasise [10]. The TWT is designed to operate at a flow rate of 15 or 60 standard cubic feet per minute, depending on the gas load at the TWT inlet.

The TWT is a computer actuated and controlled tritium re-

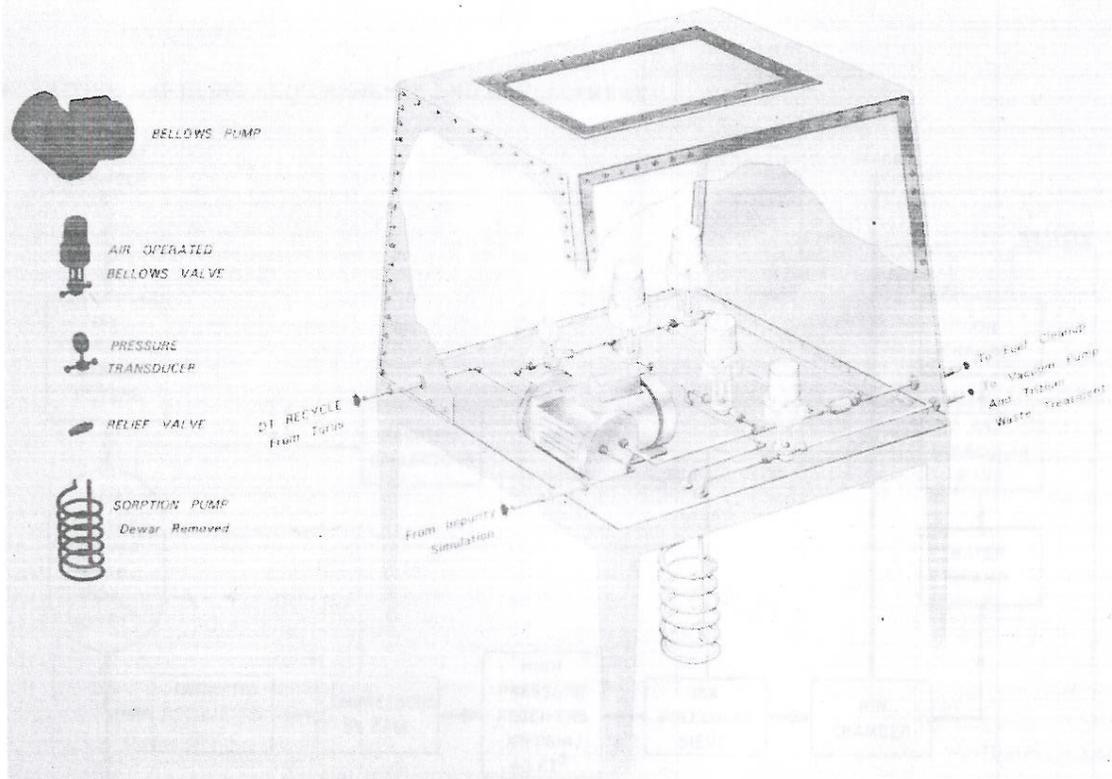


Fig. 6. Transfer pump unit enclosed in secondary containment.

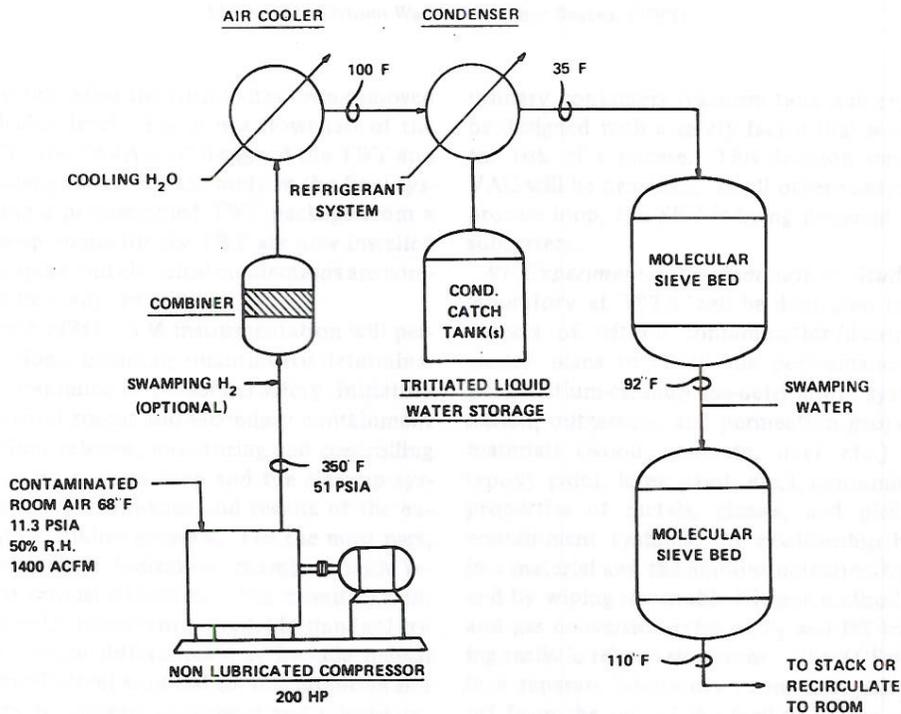


Fig. 7. The Emergency Tritium Cleanup System (ETC).

removal system that operates by the catalytic conversion of all hydrogen isotopes in the input stream to water and organic materials oxidized to water and carbon dioxide. The oxygen level will automatically be maintained in the system at levels

sufficient to ensure catalytic conversion of all hydrogen isotopes to water. The water generated by these processes will be adsorbed on a molecular sieve. Then, the remaining gaseous effluent will be discharged to the atmosphere through the

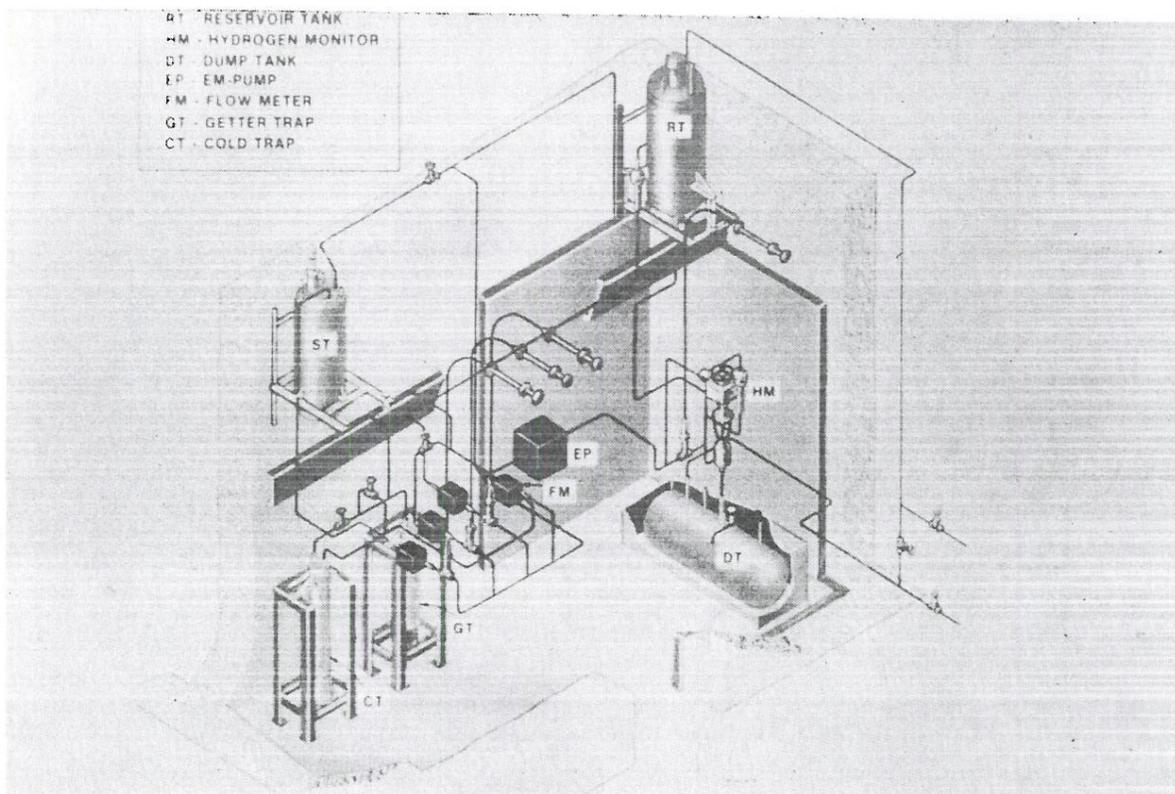


Fig. 9. The ANL Lithium Processing Test Loop (LPTL).

TSTA is designed to be a computer controlled system and will not operate fully unless MDAC is operational. Each subsystem will be able to be tested without MDAC if necessary; however, at the present time, it is anticipated that the process loop of TSTA will not operate without MDAC control. The MDAC will incorporate features to minimize potential hazards to operational personnel, the general public, and equipment. All identified hazards will be monitored and controlled by hard-wired interlocks and backed up by the monitoring of MDAC. The MDAC will monitor all radiation detectors and take appropriate action (give alarms, advise of building evacuation), if unsafe conditions are detected. The TSTA subsystems will be self-protecting to ensure against computer error resulting in a hazardous operating mode. Some equipment that may lead to hazardous situations during malfunction will have built-in absolute-limit protection to ensure against both local-manual and remote-computer errors. Checks of equipment performance will be done in software. The MDAC also will monitor input commands from critical locations to ensure that neither operations by unauthorized personnel nor errors by authorized operators will cause hazardous situations. Validation checks on computer commands will be performed in software. The computer, a Data General Eclipse, has been purchased and is installed and operating at TSTA; control software is being developed. The MDAC will use a CAMAC interface system. The computer system is operated by an Uninterruptable Power Supply, but an Emergency Generator Set will be available to operate the MDAC and critical components of the major subsystems if a commercial power loss should occur.

B. Argonne National Laboratory

The fusion reactor research program at Argonne National Laboratory includes several applied research topics covering a

variety of interrelated areas of fusion tritium technology. This work focuses principally on studies of fuel handling, breeder blanket processing, and tritium containment. These studies are both experimental and analytical in character and, in recent years, have spearheaded technological advances in many important fusion-specific areas which are described below.

Modest advances have been made in recent years in the study of liquid lithium processing. A 200-l capacity system, the Lithium Processing Test Loop (LPTL) [12] has been operating for over 9500 h (Fig. 9). Cold trapping, reactive-metal getter trapping, and a method based on molten salt extraction as a technique for removing trace impurities from liquid lithium have been tested [12] using the LPTL and related facilities. The ranges of the projected lower limit impurity control levels for the elements H(D,T), O, N, and C, based on these tests, are shown in Fig. 10. Molten salt extraction offers the best method for developing a regenerable process to recover tritium (from DT reactor blankets) and to control impurities at the <10 apm level. The salt extraction method also seems applicable to the processing of liquid-lithium-lead alloys.

In the area of instrumentation and hardware for liquid lithium systems, permeation and resistivity type meters have been developed that can be used effectively for monitoring hydrogen [13] and hydrogen plus nitrogen [13]. The principal difficulties in operating present stainless-steel lithium systems have resulted from 1) cracking of special components containing cold-worked material with high residual stress (EM-pump channels and valve bellows) [12] and 2) mass transfer buildup [12], [14] involving iron, chromium, and nickel. A method developed by DeVan *et al.* [14], wherein aluminum is added to lithium to react with and stabilize the surfaces of stainless-steel components, has shown promise as a means of retarding mass transfer and intergranular penetration in stainless-steel systems.

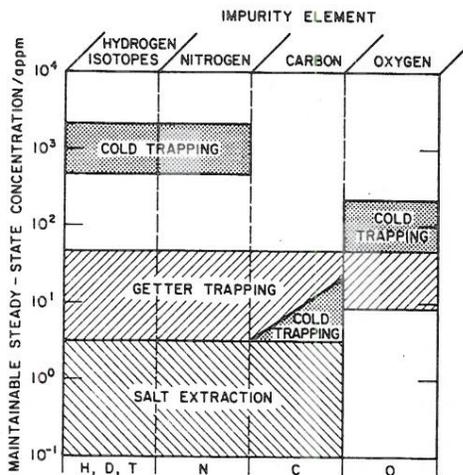


Fig. 10. Summary of projection of achievable lower-limit control levels for selected lithium processing methods.

The use of solid-lithium compounds as the breeding material for DT fusion reactors has been studied increasingly in the U.S. in recent years [15]. Three different approaches for tritium extraction have been suggested: 1) *in situ* tritium recovery, 2) removal and external processing of fuel-pin breeder assemblies on a periodic basis (every 6 to 12 months), and 3) continuous circulation of solid material into and out of the reactor with tritium processing done externally. Evidence suggests that the *in situ* recovery of tritium is feasible, but a carefully controlled breeder material temperature distribution is required [15] so that tritium does not build up to excessive levels in low-temperature regions and sintering does not occur in high-temperature regions.

The fuel-pin approach to tritium breeding has the problem of large in-blanket tritium inventories (~50 kg/fusion GW for annual pin removal), whereas the circulating-solid approach presents formidable engineering complexities. However, if the *in situ* method of handling solid breeder blankets becomes intractable and if liquid metals are eliminated from consideration because of engineering or safety considerations, the fuel-pin and circulating-solid approaches may prove to be the only recourses for self-sustaining DT fusion reactors.

In work related to the STARFIRE Commercial Fusion Reactor Study [16], at Argonne National Laboratory, a top-down selectivity analysis has been made [17] to identify the most tractable breeder/coolant/structure (B/C/S) combinations for DT fusion reactors. In a typical analysis, a breeder material (liquid-lithium, liquid-lead-lithium alloy, solid Li_7Pb_2 , Li_2O , or another lithium-containing ceramic) is matched with various coolants (water, helium, liquid metals, molten salts), and compatibility assessments are performed. The criteria for these assessments are breeding capability, exothermicity of breeder/coolant interaction, operating temperature requirements, and thermodynamic efficiency of the associated power cycle. Breeder/coolant combinations that survive the scrutiny of this first assessment are matched with likely classes of structural materials (austenitic-, nickel-, ferritic-, and refractory-base alloys) and a second assessment is made. For the second assessment, criteria are based mainly on structural material compatibility (with breeder and coolant) and temperature constraints. In summary, no B/C/S combination from the choices listed above has been developed to the point where we can confidently predict that satisfactory performance in the fusion

environment can be achieved. Liquid-lithium/refractory metal (Nb- or V-base alloys) and ceramic breeder/water (pressurized or boiling)/austenitic or ferritic combinations appear to be among the least objectionable choices from an engineering and compatibility point of view.

C. Mound Facility

The Mound Facility has been actively involved in tritium technology for over 20 years. Recently, Mound Facility has focused most of its tritium technology development on tritium containment and environmental control. The two components of Mound Facility's tritium technology development currently most active and most relevant to fusion needs are the Tritium Effluent Control Laboratory (TECL) and the Tritium Storage and Delivery System (TSDS) project for the TFTR at Princeton Plasma Physics Laboratory.

The TECL was initiated in 1971 for development and demonstration of tritium containment. The goal of TECL is to prevent any tritium release to the environment and to recover for reuse all tritium released within the laboratory. The TECL consists of an integrated set of containment systems and detritiation experiments. Tritium containment is provided by gloveboxes, a glovebox atmosphere detritiation system (GADS), a laboratory that can be isolated from the remainder of the building, and an emergency containment system (ECS). The GADS is a $0.05\text{-m}^3/\text{s}$ helium purifier that continuously cleans the glovebox atmosphere. The ECS is a $0.5\text{-m}^3/\text{s}$ catalytic oxidation, water-vapor-adsorption system for room air detritiation. In addition, a $7.5 \times 10^{-3} \text{ m}^3/\text{s}$ utility air detritiation system of the oxidation-adsorption type is available for cleaning passbox atmospheres and for nonstandard applications [18].

Four tests have measured the efficiency of the GADS and ECS in cleaning up after a tritium release in a glovebox or a room. Glovebox cleanup was as predicted; however, ECS room tests showed substantial tritium loss through imperfect exhaust duct seals. More tests are planned to study surface absorption and conversion to HTO. This was the first test of such a large ECS system in the U.S.

Tritiated water vapor collected by the ECS or by similar systems can be detritiated by the combined electrolysis catalytic exchange (CECE) pilotscale unit that is part of TECL. The CECE incorporates a countercurrent flow of water and hydrogen gas in two 2.5-cm diameter, 7.5-m-long columns packed with a hydrophobic, precious metal catalyst developed by Atomic Energy of Canada, Ltd., [19]. Bottom reflux is provided by a solid polymer electrolysis unit, whereas top reflux is provided by a catalytic recombiner. Tritiated water containing 300 Ci/l has been stripped to 10^{-3} Ci/l in the CECE at feed rates of approximately 4 ml/min of water. Although CECE has not been tested long enough to accumulate an equilibrium concentration, values in the range of 1–10 mCi/l are expected [20]. The goal of the CECE work is to develop a full-scale detritiation plant suitable for processing tritiated aqueous waste from fusion or fission reactors and from fuel reprocessing plants.

Part of the hydrogen generated in the electrolysis unit can be withdrawn to use as feed for the remaining TECL component, a cryogenic distillation system. This system includes a single 0.6-cm-diameter \times 50-cm-long packed column, operating at approximately 25 K. Bottom and top concentrations of 2500 Ci/m^3 and 10^{-3} Ci/m^3 , respectively, were measured during one run with this column, for an enrichment factor of 2.5×10^6 . Feed rates of 100 std cm^3/min are possible at this

enrichment factor. This work is supported by computer simulation studies of the distillation process. In addition, an experimental study of the hydrogen isotope equilibration rate ($2DT \rightleftharpoons T_2 + D_2$) at cryogenic temperatures is being performed. This study is of interest both in cryogenic distillation and in cryogenic fuel-pellet production.

The Tritium Storage and Delivery System (TSDS) has been designed and constructed for use in the TFTR at the PPPL. The TSDS will receive, assay, store, and deliver measured quantities of high-purity tritium to fuel the TFTR. The TSDS consists of a receiving manifold, uranium tritide storage beds, transfer pumps, and metering volumes, all contained in two stainless-steel gloveboxes. In addition, a quadrupole mass analyzer and pressure-volume-temperature measuring equipment permit assay of the tritium as received, as well as confirmation that the tritium delivered to the reactor has the required purity. Extensive development has done with the mass analyzer to permit analysis of the required sensitivity and accuracy [21].

In operation, tritium received at TFTR is introduced to the TSDS through the receiving manifold; after being assayed, it is pumped onto a uranium bed where it is stored as uranium tritide for future use. During DT operation of TFTR, an appropriate tritium storage bed is heated to raise the internal tritium pressure to about 0.5–1 atm. Each time the reactor is to be fueled (as often as every 5 min), the required amount of tritium is pumped into a metering volume. The gas then passes through a delivery manifold to three calibrated injection volumes near the torus. Finally, injection to the torus is controlled by a specially designed piezo electric valve at each injection volume. After a burn, the fuel is not recycled but is recovered by the TFTR vacuum system and stored for later reclamation.

The TSDS is designed to achieve the highest possible levels of reliability, safety, and tritium containment. The two transfer pumps are interchangeable two-stage doubly contained metal-bellows pumps, one of which is a spare. The three interchangeable storage beds are each doubly contained in stainless steel, with a provision for purging the secondary volume of tritium that permeates from the primary container. Each bed has a rupture-disk-sealed connection to an evacuated "dump" tank. At any given time, one storage bed is active, one is available for cleaning operations, and one is a spare.

Complete instrumentation is provided for TSDS to permit accurate control and to promote safety. The tritium generators are provided with pressure sensors, redundant thermal control, and overtemperature protection.

Normal operations of the TSDS are performed remotely through computer control of pneumatically operated valves. All process valves are bellows sealed with polyimide instead of metal. These valves have been remachined individually to ensure reliable leak-free operation.

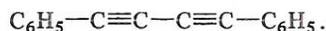
The TSDS will be tested with a load of 10^4 Ci of tritium to ensure reliable operation of the entire system before delivery to the Princeton Plasma Physics Laboratory. For this test a dummy manifold and injection chamber will simulate the manifold and torus at Princeton.

D. Lawrence Livermore National Laboratory

Several areas of tritium technology of interest to the fusion energy community are being pursued at the Lawrence Livermore National Laboratory. One area of concern is the tritium recovery following an environmental or atmospheric release of tritium. In current designs of systems for recovery of such released tritium, the method of recovery is catalytic oxidation

with atmospheric oxygen forming tritiated water, which then is collected. Sherwood [22] has measured room temperature kinetic data on tritium/air oxidation with three common catalyst/substrate formulations. These three formulations were platinum/alumina, palladium/kaolin, and palladium/zeolite. Compared to self-catalyzed atmospheric conversion, each of the dispersed-metal catalysts is extremely effective in promoting tritium oxidation; equivalent first-order rate constants are higher by roughly nine orders of magnitude. Electron-microprobe scans reveal that the dispersed metal is deposited near the outer surface of the catalyst, with metal concentration decreasing exponentially from the pellet surface. The platinum-based catalyst is more effective than the palladium catalyst on a surface-area basis by about a factor of three.

Miller *et al.* [23] at Lawrence Livermore National Laboratory are working on the development of an organic getter which will operate in the presence of air and will minimize the formation of the more hazardous tritiated water. The compound of interest, 1,4-diphenylbutadiyne, is a hydrocarbon,



The T_2 adds to the acetylene (triple) bond of the getter in the presence of a metal catalyst. However, the necessary catalyst will stimulate the $T_2 + O_2$ reaction, so some tritiated water will be formed. Early results indicate that these getters will indeed remove tritium from air, although with the formation of some water. The organic triple bond appears to be a very good candidate for this type getter. Miller states that the best solution would be to combine the metallic catalyst and the triple bond in a single fairly simple molecule, for instance $(PhC\equiv CPh)_2Pt$, where Ph is a phenyl group. To date, very little work has been done on methods of disposing of or storing these tritiated organic getters. This work will be done when an optimum getter material is chosen, and the gettering process is thoroughly understood.

The Rotating Target Neutron Source (RTNS-II) at Lawrence Livermore National Laboratory [24] is an accelerator-based neutron source for studying radiation damage to materials. Energetic deuterons bombard a solid metal tritide target, producing fusion neutrons (over 10^{13} 14-MeV neutrons/s). Deuterium continually displaces tritium from the target at rates as high as 7 Ci/h. The anticipated addition of a second accelerator and increased neutron yield of the present accelerator could increase tritium output to 20–30 Ci/h. This tritium is released into the accelerator vacuum system. Because it is not acceptable to release this tritium to the environment, a tritium-scrubbing system was devised [25] to clean the vacuum system exhaust before venting it to the atmosphere. This system consists of a catalytic recombiner, where tritiated water is formed, and molecular sieve drying towers for collecting the water. When these molecular sieve driers become saturated, they are replaced and the saturated beds are buried. The driers contain approximately 7 lbs of molecular sieve loaded to 14–15 percent of the dry weight. At current operating levels, water loads of ~ 0.25 lb/week are collected. Most of this is D_2O , with only ~ 1 percent being tritium. Schumacher reports a 10^5 – 10^6 tritium concentration reduction factor through the scrubber system, with most of the escaping tritium in the gaseous form. Small scrubber systems, such as the one at RTNS-II, may find extensive use in the fusion program.

Souers [26], at Lawrence Livermore National Laboratory, is measuring and correlating cryogenic data on D_2 , T_2 and mixtures of these components. Because solid DT may be a future

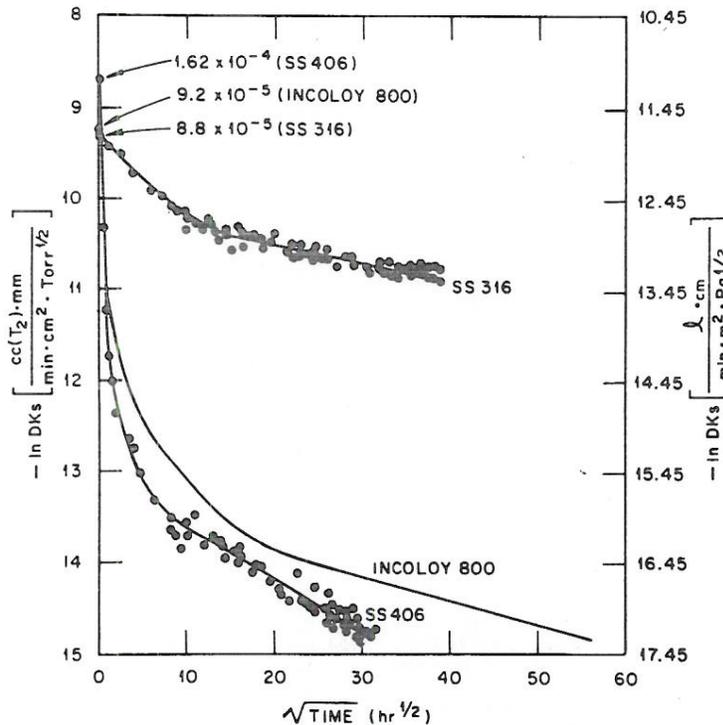


Fig. 11. Tritium permeabilities of three construction alloys at 930 K as the alloys are oxidized by water at 0.94 atmosphere.

fusion fuel, the measurement of physical and chemical properties of cryogenic DT in the solid, liquid, and gas phases will aid the design engineer and the plasma physicist considering the use of cryogenic DT.

Some of the properties being measured include the DT reaction rate, that is, the rate at which $D_2 + T_2$ react to form the three-component mixture D_2-DT-T_2 . At room temperature, this reaction takes place with an exponential $1/e$ -time on the order of tens of minutes. At 20 K, the $1/e$ -times are on the orders of tens of hours. Other properties being measured by Souers include thermal conductivity of the solid mixtures and electrical conductivity of both liquid and gaseous species in the 20–26 K range. Souers has compiled a very important report [27] in which he correlated the measured physical and chemical properties of the hydrogens <30 K. This compilation will benefit fusion engineers who do not have a strong background in cryogenic materials, as well as those specialists who will be concerned with details of the low-temperature DT mixtures.

E. Oak Ridge National Laboratory

Bell *et al.* [28] have measured the tritium permeability of structural materials and surface effects on permeation rates. Tritium management in any system always should include containment so that tritium release rates will be less than established limits and established limits will be as low as practicable. The ability of hydrogen isotopes to permeate most materials makes complete tritium containment an extremely difficult task. However, tritium release rates from a given system can be minimized by two primary efforts. First, the selection of a compatible containment material which, frequently, also will be the structural material. This effort would include finding films and barriers compatible with the structural material. Second, and sometimes more important, the material should

have surface chemistry that continuously impedes tritium permeation. Bell has shown that by exposing the austenitic Incoloy 800 and the ferritic SS 406 to steam oxidation at 0.94 atm and 930 K, the permeabilities of these materials can be reduced by several orders of magnitude over the permeability of the clean metal (Fig. 11). These studies demonstrate that *in situ* surface oxidation of construction alloys can produce oxide barriers that reduce tritium permeation by significant factors. However, these results have limitations, when applied by extrapolation, to operating systems with high temperatures and corrosive conditions. Therefore, Bell emphasizes [28] that current conclusions of permeation barrier effects must be considered only as indications of effects under actual operating conditions. This type of experiment needs to be repeated on a larger scale where conditions more nearly match reactor type conditions. The demonstration that, indeed, these permeation barriers can be formed *in situ*, under reactor conditions, would be a very important development for the fusion energy program.

III. CONCLUSIONS

Although this discussion shows the breadth of tritium technology development in the U.S., it cannot begin to cover all the many research and development programs in the field. We hope to show that there is a large, serious effort underway to develop this technology in a timely manner. In the current programs, tritium technology should not be an obstacle to the design and construction of the FED. These continuing programs will develop the data base and experience necessary to proceed from conceptual design to engineering design and construction of the FED.

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