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TRITIUM PROCESSING AND CONTAINMENT TECHNOLOGY FOR FUSION REACTORS: PERSPECTIVE AND STATUS

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TRITIUM PROCESSING AND CONTAINMENT TECHNOLOGY FOR FUSION REACTORS: PERSPECTIVE AND STATUS*

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This paper reviews the status of selected tritium processing and containment technologies that will be required to support the development of the fusion energy program. Considered in order are the fuel conditioning and recycle systems, the containment and cleanup systems, the blanket processing systems, and two unique problems relating to tritium interactions in neutral beam injectors and first wall coolant circuits. The major technical problem areas appear to lie in the development of (1) high-capacity, rapid recycle plasma chamber evacuation systems; (2) large-capacity (≥100,000 cfm) air handling and processing systems for atmospheric detritiation; (3) tritium recovery technology for liquid lithium blanket concepts; (4) tritium compatible neutral injector systems; and (5) an overall approach to tritium handling and containment that guarantees near zero release to the environment at a bearable cost.

INTRODUCTION

The tritium handling and containment requirements of both near-term and longer-range fusion devices have been under study for nearly a decade. Although still in its infancy, the tritium technology program that is evolving in support of the development of fusion power is beginning to make substantial progress in terms of (1) defining the criteria for fusion reactor tritium facility operations and (2) identifying tractible design solutions in a number of eminent problem areas. By coupling this progress with the rather sizeable tritium technology base that already exists as a result of other tritium related programs at Savannah River Laboratory, Mound Laboratory, Lawrence Livermore Laboratory, Los Alamos Scientific Laboratory and other research and development centers, it is now possible to define in broad terms the scope of much of the work that will be required to fullfill the tritium handling and containment requirements for the first generation of DT burning fusion devices and for those that lie beyond.

The purpose of this paper is threefold: (1) to provide some perspective on the nature of the tritium processing and containment problems currently envisioned for DT burning fusion devices, (2) to identify the areas where major research and development efforts will be needed, and (3) to highlight some of the more significant advances of recent years in terms of criteria development and solution of problems. Although many of the discussions contained herein are conjectural in nature and

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representative of the author's view, a great deal of the thought and virtually all of the factual information were culled from the references accompanying this paper and from private communications to the author by knowledgeable members of the controlled thermonuclear research and tritium technology communities wi* .. the USA. The discussions following this introduction contain summaries of the technological requirements associated with (1) the mainstream fuel cycle (with emphasis on DT burning devices), (2) containment and cleanup systems, (3) blanket processing systems, and (4) several unique problem areas associated with systems that interface with the tritium handling systems. Table 1 contains an outlined summary of research and development items for each of the discussion areas. Table 2 contains a summary of the focus of work in selected ongoing research and development activities within the USA that is directly su. ortive of or relevant to the fusion tritium technology program.

THE MAINSTREAM FUEL CYCLE

The most important near-term tritium handling problems, and perhaps the most neglected to date, are those that involve the mainstream of the fuel cycle for DT burning fusion reactors. Because only a few percent of the fuel delivered to the plasma chamber is actually consumed during a typical burn cycle (tokamaks, mirrors, or theta-pinches), it is absolutely essential that the unburned fuel be recycled from an economic viewpoi t alone. The principal functions of the mainstream fuel recycling system are (1) to provide for evacuation of the plasma chamber in a way that permits accumulation and consolidation of all the unburned fuel in an easily recycleable form; (2) to reduce particulate debris (from plasma wall interactions) and non-hydrogenous impurity element concentrations in the fuel to levels that are acceptable for refueling purposes; (3) to remove protium (^1H) from the D-T mixture and to adjust the D/T ratios to values required for direct refueling, energetic neutral injection, pellet fueling, etc.; and (4) to provide the means for circulation, compression, adjustment of physical and chemical state, and interim storage of the fuel. A summary discussion of each of these functions is given below; more comprehensive discussions may be found in references 1 through 8.

Plasma Chamber Evacuation Systems

Several recent studies[1-5] have provided some perspective on the evacuation requirements for experimental tokamak reactors. Clearly, there will be need to handle large gas loads (on the order of 5 Torr-liters per thermal MW) in short time periods and to do so repetitively with little or no interruption. The pumping equipment must be reliable and maintainable, must operate in a high radiation environment, must be capable of pumping all atomic and molecular species present in the plasma chamber following a burn cycle, must be sufficiently compact to be accommodated by the reactors requisite physical configuration, and must be reasonably economic in terms of capital and operating costs. Of course, the pumping systems must also be compatible with and must provide a high degree of containment for large quantities of tritium.

Three basic types of high-speed, large-capacity pumping methods have been considered for use in near-term experimental reactors[1,2,4]: (1) dynamic evacuation using diffusion or turbomolecular pumps, (2) gettering with active metals such as zirconium aluminum alloy or titanium metal, and (3) cryogenic evacuation using cryocondensation or cryosorption pumps. The dynamic methods

TABLE 1. Tritium Related Research and Development in Support of the Fusion Energy Program

I. Fuel Conditioning and Recycle
   A. Plasma Chamber Evacuation Technology
      1. Identify and develop high-capacity, rapid recycle evacuation methods.
      2. Verify compatibility with tritium and adequacy of tritium containment.
      3. Establish interface-technology with plasma chamber and with regeneration systems.
      4. Determine consequences of radiation effects and maintenance requirements.
   B. Impurity Removal and Monitoring
      1. Analyze debris transport mechanisms and determine consequences.
      2. Develop debris separation and handling technology.
      3. Develop nonmetallic element removal methodology.
      4. Develop helium removal methodology.
      5. Establish impurity monitoring methods.
   C. Hydrogen Isotope Enrichment
      1. Define separations and enrichment needs.
      2. Identify usable enrichment technology.
      3. Determine optimum processing modes (batch vs. continuous).
      4. Develop low temperature isotopomeric equilibration methods.
      5. Establish cost/benefit factors affecting alternative enrichment strategies.
      6. Proof test instrumentation and control systems for enrichment assemblies.
   D. Hardware Development
      1. Establish criteria for hardware performance.
      2. Identify hardware development needs (e.g., valves, compressors, pumps, traps).
      3. Verify compatibility with tritium and adequacy of tritium containment.
   E. Integrated Systems Tests (Fuel Cycle Simulation)
      1. Verify identification of all processing steps.
      2. Optimize processing sequence.
      3. Determine hardware and component interfacing requirements.
      4. Conduct fuel cycle simulation studies.

II. Containment and Cleanup Systems
   A. Primary Containment
      1. Establish optimum structural materials (e.g., permeation resistant, nonembrittling).
      2. Develop leak-free assembly technology for permanent and temporary connections.
      3. Develop permeation barrier methods (e.g., coatings, composites) for both ambient and elevated temperature operation.
      4. Investigate permeation rates and mechanisms at low driving pressure.
   B. Secondary Containment
      1. Identify secondary containment requirements.
      2. Develop and test peripheral jacketing methods.
      3. Determine optimum flow patterns and geometries.
      4. Establish access and maintenance requirements.

TABLE 1 (Cont'd.)

5. Analyze features of permanent and portable enclosures (e.g., gloveboxes, fumehoods).
6. Develop purge processing technology.

C. Tertiary Containment

1. Develop and test high-velocity air circulation and processing systems.
2. Identify, characterize, and optimize oxidation catalysts, adsorber beds, thermal economizers, and regeneration procedures for air detritiation systems.
3. Determine effects of major variables (e.g., temperature, humidity, flow pattern, air mixing).
4. Develop methods to minimize residual contamination and outgassing following releases (e.g., short cleanup times, hydrophobic coatings).
5. Investigate kinetics of reactions involving tritiated species in ambient atmospheres.
6. Establish cost/benefit factors for alternative cleanup strategies.

D. Waste Disposal Technology

1. Develop strategies and methodology for maximum tritium recycle.
2. Establish technology for safe, compact, low long-term release tritiated waste disposal.

E. Integrated Systems Tests

1. Verify integrated operation of the three levels of containment (primary, secondary, and tertiary).
2. Carry out realistic deliberate-release experiments.

III. Storage, Shipping, and Safeguards

A. Storage

1. Develop and test fuel storage concepts.
2. Establish criteria for storage vault integrity.

B. Shipping

1. Establish standards for shipping and verify compliance with federal regulations.
2. Verify production capabilities.
3. Examine anticipated product and shipping costs.
4. Determine production scheduling requirements and concomitant programmatic impacts.

C. Safeguards

1. Establish safeguards implications for facility and shipping operations.
2. Determine reactor surveillance and site protection criteria.
3. Prepare safeguards planning logics for near-term and longer-range devices and facilities.

IV. Blanket Processing Technology

A. Liquid Lithium Blanket Concepts

1. Establish criteria for tritium containment, inventory, and recovery.
2. Identify and develop blanket processing methodology.
3. Verify maintainability and reliability of blanket processing systems.

B. Solid Blanket Concepts

1. Verify adequacy of tritium release rates from solid blanket materials.
2. Establish long-term performance of solid blanket materials in anticipated radiation environments.
3. Develop and test tritium

TABLE 1 (Cont'd.)

recovery methods.

C. Molten Salt Blanket Concepts
   1. Determine chemical effects of transmutation reactions that produce tritium.
   2. Develop and test tritium recovery methods.
   3. Verify maintainability and reliability of blanket processing systems.

V. Selected Additional Topics

A. Instrumentation and Control Systems
   1. Develop fuel cycle diagnostic systems.
   2. Develop breeder blanket diagnostic systems.
   3. Establish integrated tritium facility maintenance, control, and response strategies and develop systems.
   4. Verify performance of monitoring and diagnostic methods in anticipated power reactor radiation environments.

B. Neutral Beam Injector Interfacing
   1. Identify important tritium interactions in neutral injector systems.
   2. Establish tritium containment criteria and develop containment methods.
   3. Determine impact of tritium containment on injector maintenance.

C. Coolant/Tritium Interactions
   1. Identify and characterize modes of tritium insertion into coolant systems.
   2. Establish tritium buildup rates for hydrogenous coolant concepts.
   3. Develop methods for removal of tritium from nonhydrogenous coolants.
   4. Examine effects of energetic tritium implantation (first wall).
   5. Examine effects of radiolysis in tritium contaminated coolants.
   6. Determine the mechanisms for, and magnitudes of, tritium releases from coolant circuitry.

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generally require fore-pumping and/or fore-collection systems, are large in size, and may be questionable in terms of tritium compatibility and containment. The getter and cryogenic methods are committed to regenerative operation; hence, where continuous on-line pumping is necessary there would undoubtedly be need for paralleling redundant systems to permit simultaneous pumping and regeneration.

In terms of all of the above considerations, cryosorption pumping at 4 K appears to offer the best prospects for meeting plasma chamber evacuation requirements in both near-term and longer-range tokamak reactors and in related devices with similar evacuation needs. This contention is generally well supported by the recent studies of Watson and Fisher[9,10] and by the earlier work of Stern et al.[11,12]

Impurity Removal

The principal impurities identified thus far that will in all likelihood be present in the plasma exhaust of magnetic-confinement fusion devices are fine particulate solid debris and the nonmetallic elements He, O, N, and C. Considerations of the problems associated with the removal of each type of impurity are summarized below.

The harsh radiation environment anticipated for the first wall of a fusion reactor is likely to lead to a dislodging of sizeable quantities of fine particulate metal and metal compounds. Although most of this debris is expected to either settle

TABLE 2. Summary of Selected Tritium Research and Development Activities in Support of the Fusion Energy Program

Research and Development Area	Laboratory and Cognizant Personnel	Nature of Work	References
Mainstream Fuel Cycle	ORNL (J. S. Watson, P. W. Fischer, S. D. Clinton et al.)	Development of Cryopumping Technology Analysis of Fuel Cycle Strategies	4, 9, 10, 35
	LLL (R. G. Hickman, T. R. Galloway, V. P. Gede et al.)	Fuel Cycle Strategies for Minor Devices Tritium Gettering and Storage Materials and Hardware Assessments	7, 17, 18, 19, 27
	ML (L. J. Wittenberg, W. R. Wilkes et al.)	Surveys of Existing Technology Hydrogen Isotope Separations Metal Tritide Technology	3, 13, 14, 15
	ANL (V. A. Maroni, B. Misra et al.)	Analysis of Fuel Cycle Strategies Analysis of Enrichment Strategies	1, 2, 16
	LASL (J. L. Anderson, R. H. Sherman et al.)	Fuel Cycle Strategies for θ-Pinches Fuel Recycle in the INS	8, 51
Containment and Cleanup Systems	LLL (T. R. Galloway, A. E. Sherwood, M. F. Singleton et al.)	Analysis of Cleanup and Containment Strategies Tritium Oxidation, Adsorption, and Gettering Tritium Containment in Laser Fusion Tritium Containment in the RTNS	7, 27, 32, 33
	ML (W. R. Wilkes, L. J. Wittenberg, E. A. Mershad, J. Kershner et al.)	Experiments with Air Detritiation Systems Analysis of Cleanup and Containment Strategies Tritium Waste Treatment Tritium Effluent Control	3, 13, 28, 29, 31
Blanket Chemistry and Blanket Processing Technology	LASL (J. L. Anderson, D. Carstens et al.)	Studies of Liquid Metal Alloy Extraction Concepts for Processing Liquid Lithium	39
	ORNL (J. S. Watson, J. B. Talbot, J. T. Bell, F. J. Smith, G. M. Begun et al.)	Tritium Removal from Lithium Alloys Tritium Sorption from Liquid Metals Thermodynamic Studies of Hydrogen Isotope Solutions in Lithium and Lithium Alloys	35, 36, 44
	BNL (J. R. Powell, R. H. Wiswall et al.)	Analysis of Breeder Blanket Strategies Experimental Studies with Solid Blanket Materials	40
	ANL (W. F. Calaway, E. Veleckis, V. A. Maroni et al.)	Studies of Molten Salt Extraction Concepts for Tritium Removal from Liquid Lithium Thermodynamic Studies of Hydrogen Isotope Solutions in Lithium and Lithium Alloys Development of Lithium Processing Technology	37, 38
	Univ. of Wisconsin (E. M. Larsen, R. G. Clemmer, D. K. Sze et al.)	Development of Liquid and Solid Blanket Design Concepts	41, 45

TABLE 2 (Cont'd.)

Research and Development Area	Laboratory and Cognizant Personnel	Nature of Work	References
Hydrogen Isotope Permeation and Other Physicochemical Studies	LLL (P. C. Souers, J. W. Pyper, R. G. Hickman et al.)	Physicochemical Studies of Deuterium Tritide (DT) Tritium Implantation Effects	7, 49, 50
	Princeton Univ. (R. C. Axtman, E. F. Johnson, H. K. Perkins et al.)	Hydrogen Permeation at Low Pressures Tritium Holdup Due to Coatings Chemical Engineering Analyses of Tritium in Molten Salts	21, 22, 43
	ANL (E. Van Deventer, V. A. Maroni, B. Misra et al.)	Development of Permeation Barriers using Multiplex Materials Kinetics of Reactions Involving Tritiated Species	23, 38
	ORNL (J. T. Bell, S. D. Clinton, J. D. Redman, F. J. Smith et al.)	Tritium Permeation through Steam Generator Materials Oxidation of Permeating Tritium Sorption Pumping by Deep Beds	20, 35
	Sandia Livermore (W. A. Swansiger et al.)	Hydrogen Isotope Permeation through Metals	25
	N. C. State Univ. (T. S. Elleman et al.)	Tritium Diffusion in Metals and Ceramics	24

out or reattach itself to the first wall, plans must be made to accommodate the migration of some particulate material into the vacuum pumps and beyond them. Because the first wall will become highly radioactive after only a few days of operation even at modest wall loadings (0.2 to 1.0 MW/m^2), the debris will also be highly radioactive and maintenance of the vacuum pumps and the equipment immediately downstream of them will be subject to increased complexity. Large quantities of debris in the torus exhaust gases could have an adverse effect on pump lifetime and performance, but any attempt to microfilter the debris in advance of the vacuum pumps (between the plasma chamber and the pumps) would more than likely lead to unacceptably large conductance losses. During cleaning or regeneration of the vacuum system, the escaping gases will probably fluidize some of the debris that entered the pumps during the torus pumping cycle. (Hopefully, most of the debris will indeed carry through the vacuum pumps so that pump lifetimes can be extended.) Removal of this debris from the fuel recycle stream can probably be done with some combination of electrostatic precipitators or millipore filters. If possible, the debris separation system should be located immediately downstream of the cryopumps (to the extent that this location does not adversely extend regeneration times). It is difficult to make accurate predictions of the particle size, size distribution, and quantity of debris generated by a prototypal fusion device; hence, some experimentation will ultimately be required in order to test debris removal methods in flowing hydrogen streams that simulate the mainstream of reactor fuel cycles.

The perpetual presence of helium,

oxygen, carbon, nitrogen, and other nonmetallic impurities in the torus exhaust of prototypal fusion devices will, for all practical purposes, be unavoidable. In keeping with the concept of fuel recycle, and in order to reduce impurity levels in the preburn fuel mixture as far as is possible, it will be necessary to provide for continuous removal of nonmetallic elements in the plasma exhaust. This should certainly be done in advance of the isotopic enrichment step and perhaps done again in advance of fuel storage so that the capacity of the storage material is not reduced by reaction with impurities.

Impurity removal can probably be carried out in a relatively straightforward manner using appropriately selected catalytic and/or getter-type beds designed to (1) crack water, hydrocarbons, and other hydrogenous compounds and (2) actively remove the impurities by reaction to form stable, nonvolatile, nonhydrogenous compounds. If the getter bed is also employed to clear the plasma chamber evacuation system, it should be designed to sorb and release hydrogen over a relatively narrow temperature range, should be reasonably compact, and should be readily disposable. It is expected that these getter beds will have to reduce O, N, C and related nonmetallic impurities in the fuel stream to the sub-ppm range. The helium present in the cryopump exhaust either could be allowed to carry over into the enrichment system and be removed therefrom as an inert/noncondensible phase (providing this carryover does not compromise the enrichment operation) or could be separated from the hydrogen isotopes in advance of enrichment using a permeable window. In addition to the need to identify and test getter bed materials (or combinations of materials) for broad spectrum nonmetallic impurity removal, the relative merits of continuous versus various degrees of batch processing of the plasma exhaust remain to be evaluated. References 13 and 14 contain surveys and discussions of existing tritium gettering and storage experience.

Isotopic Enrichment

Isotopic enrichment of the hydrogen isotopes in the mainstream of the fuel cycle will be needed to (1) reduce protium to acceptable levels, (2) adjust the D/T ratio of the bulk fuel to values prescribed by the cold fueling criteria, and (3) separate isotopically pure D and/or T streams for energetic neutral injection where this method of plasma heating is employed. Although not necessarily a part of the mainstream fuel cycle, there may also be need of a capability to provide for enrichment of tritium from high-level wastes containing large relative amounts of protium. The methods employed for each of the above isotopic enrichment processes in DT burning reactors will depend to a large extent on the relative isotopic concentrations involved, the magnitude of separation to be achieved, the quantity of fuel to be enriched, and the total amount of tritium to be handled. Although a variety of methods have been identified which could be applied to hydrogen isotope enrichment (including cryogenic distillation, chromatography, electrolysis, laser stimulated separations, and thermal diffusion), the matching of method with application will undoubtedly depend on economy and reliability. Considering the large spent fuel flow rate for even experimental scale reactors (e.g., several kilograms per day of DT for tokamak experimental power reactors[1,4]), cryogenic distillation appears to be the most practical, reliable, and economic method for mainstream enrichment.

A number of recent studies have addressed the question of hydrogen isotope enrichment for near-term fusion devices[1,4,15,16]. The general conclusions of these studies have been that cryogenic distillation methods would permit the required enrichments to be made for most currently conceived devices, including totally driven DT reactors where a nearly complete separation of the deuterium and tritium in the plasma exhaust must be made. These separations generally require from 3 to 6 columns with the number of theoretical stages per column ranging from 30 to 50. One or more equilibrators to re-establish or alter the isotopomeric equilibria among the six isotopomeric forms of hydrogen (H_2, D_2, T_2, HD, HT, DT) is also usually needed, depending on the magnitude of H/T and/or D/T separation required. Power input levels to drive the distillation cascade (including refrigeration and compression power) are found to be only a small fraction of the total plant recirculating power (i.e., <<1%).

Fuel Storage

The identification of fail-safe fuel storage methodology will be essential to the development of a minimum credible impact tritium handling facility for DT fueled fusion power plants because, in all probability, the major fraction of the tritium contained within such plants will be in the storage reservoir. Although a comprehensive design basis remains to be established for these storage systems, it is reasonable to assume that they will be disaster proof, barricaded vaults containing parallel and series arrangements of storage cells constructed in such a way that failure to operate of one or perhaps several cells does not constitute an unresolvable maintenance problem. Plausible concepts for the storage cells include (1) compressed gas cylinders, (2) thermally regenerative metal hydride beds, and (3) cryostated liquid or solid hydrogen storage tanks. Of these, metal hydride storage concepts are generally regarded as being best suited to the storage of large quantities of tritium. The storage vaults would have at least three principal access requirements: (1) reception of purified and appropriately enriched recycle fuel, (2) reception of incoming tritium from tritium production facilities, and (3) release of stored fuel to the fuel blending and delivery systems. References 1 through 7, 13, 14, and 17 discuss some strategies and methods for fuel storage that would be applicable to most fusion reactor concepts.

Materials and Hardware

The selection, qualification, and specification of materials, fabrication practices, and hardware items required to assemble fuel handling and transfer systems will be an integral part of the fuel cycle development program for coming generations of fusion devices. Included in this phase of the program would be consideration of (1) structural materials used in transfer lines, valves, pumps, compressors, vessels, secondary containment systems, and related hardware; (2) elastomers, gasket and sealing materials, lubricants, coating substances, paints, and other materials employed in nonstructural applications; and (3) the general hardware methodology to be applied in tritium transfer and compression operations.

For applications at ambient temperature, the 18-8 stainless steels are regarded[18] as being qualified for use in a high level tritium environment. Welding is the preferred joining method, but junctions employing metal gasket seals could be used where removable couplings are needed to connect integrated sections of the fuel cycle. It is generally recommended that joining methods which employ polymer seals be kept to a minimum and be restricted to low level tritium environments[3,18]. Threaded seals are not considered to be suitable for tritium service and should be avoided.

It appears advisable that hardware items such as pumps, valves, or compressors which require internally moving parts, nonetheless be designed to have fully welded structures rather than hermetric or fluid seals. Where rotating or sliding seals are unavoidable, the use of techniques employing magnetic-coupling or concentric bellows drives is recommended. All hardware items that are not fully welded will probably require a secondary containment shell with a processible atmosphere. Valves with all-welded bellows-type construction are available for use at gas pressures ranging from hard vacuum to 2500 psi and at temperatures up to 350°C. Valves of this type that are adequate in size for most fuel cycle applications are available; but large, high conductance gate-type valves suitable for use at the vacuum system/plasma chamber interface will require some development. Several types of fully welded diaphragm and bellows-type pumps and compressors have been used successfully for tritium transfer operations[19] but the throughput rates could stand considerable increase. In many cases the long-term performances of these and related hardware items needs to be examined under the conditions of radiation environment, magnetic field, and temperature that are anticipated for their use in fusion devices. References 3, 13, 18, and 19 contain discussions of criteria for materials and hardware selection, component design definition, and fabrication methodology in tritium systems.

CONTAINMENT AND CLEANUP SYSTEMS

Accepting the notion that extremely strict tritium containment guidelines will be set for future fusion power plants and, in the light of past and current experience with tritium containment in ongoing programs at laboratories in the USA and worldwide, it seems prudent to consider at least three levels of containment in fusion reactor facilities. For the purposes of this paper, these three levels will be identified as primary, secondary, and tertiary; where primary refers to the construction materials and components in direct contact with tritium, secondary refers to the various local containment enclosures surrounding major tritium handling hardware and components, and tertiary refers to the large-volume room air handling systems.

Primary Containment

As a part of the primary containment level, consideration would be given to the identification of materials, hardware, and fabrication methods that offered (1) a high degree of protection against leakage and permeation, (2) high component reliability with minimum maintenance, and (3) adequate containment without an excessive cost burden. Among the materials selection criteria would be the requirement that primary materials be resistant to embrittlement and other forms of deterioration that can occur in a tritium environment. Resistance to permeation by tritium will be important, particularly in elevated temperature (>300°C) structures. This will undoubtedly result in the need for special permeation barriers such as multiplex materials, ceramic surface coatings, and other related preparations. There have been numerous studies in recent years concerning the nature of hydrogen permeation under anticipated fusion reactor conditions. A series of reports[20-25] that provide some perspective as to the general scope and direction of currently ongoing programs is included in the references to this paper.

Secondary Containment

The secondary containment systems will consist of integral jackets and other close fitting enclosures around transfer lines, valves, and other fuel cycle hardware. Gloveboxes, fumehoods, and portable enclosures would also be utilized to house entire tritium facility systems that require hands-on operation or maintenance at regular intervals. In particular, portable enclosures would be employed during all routine and off-normal maintenance operations where a potential for tritium release exists. In all cases, the atmosphere in these secondary containment systems (be it air, nitrogen, argon, helium, etc.) would be totally separated from breathing air systems, would be monitored continuously, and would be processed to remove tritium on whatever schedule is necessary to maintain breathable air conditions in the reactor or tritium facility buildings. References 3 and 13 contain some general information on the scope of and experience with secondary containment systems.

Tertiary Containment

Tertiary containment applies to the air handling and detritiation systems servicing the reactor hall and other facility rooms where potential for either entry or egress of tritium exists. Preliminary assessments have shown that the requirements imposed by the need to provide large-scale atmospheric detritiation in even experimental scale reactor buildings are a major concern from the standpoint of (1) maintenance access during reactor down periods, (2) compactness of tritium handling equipment, (3) limitations to the spread of tritium contamination, and (4) overall tritium facility costs. During the past year, several studies[1,3,27,29] have addressed the scope of these requirements with respect to gas handling rates, exigencies of the cleanup schedule, and the dominant features of the cost/benefit algorithm. The essence of these studies is summarized below.

Consider a large reactor hall of volume, V_{TOT}, having a baseline tritium level, N°. Assume that the room air is processed at a volumetric flow rate, $\dot{V}$, and that the processor efficiency is ε. The rate of tritium removal is given by

$$\frac{dN}{dt} = \frac{\varepsilon \cdot \dot{V} \cdot N}{V_{TOT}} \qquad (1)$$

The amount of time, t, it takes to reduce a massive tritium release from the maximum value following the incident, N', back to N° is obtained by integrating Eq. (1) to produce

$$\ln \frac{N'}{N^\circ} = \varepsilon \frac{\dot{V}}{V_{TOT}} t \qquad (2)$$

Values of $\dot{V}$ for selected values of N'/N°, ε, and t are given in Table 3 for a room with $V_{TOT} = 10^7$ cu. ft. ($\sim$ EPR size)[1].

According to Engelhard Industries[26], the largest unit they have evaluated to date called for 6×10^3 cfm at a cost of $\sim 10^6$ \$ for the equipment alone. Because the tritium released to the hall will rapidly soak into the surfaces of the reactor hardware and the building itself, it is advantageous to have the capability for cleaning up spills within hours after a release. If, for example, the limit is set at $\lesssim$5 hours, then the $\dot{V}$ requirements would be $\sim$10 times those for the two-day case in Table 3 and would be 100 times greater than the maximum size unit upon which Engelhard has made a quote.

A second approach to the massive release problem might be rapid cyclic flushing of the reactor hall by alternately compressing its contents (reducing room pressure by a factor η) and backfilling with clean air or an alternative cover gas. If the compressed

TABLE 3. Analysis of Recycle Flow Scenario

Amount Released	$\left(\frac{N'}{N^\circ}\right)^{(a)}$	$\ln\left(\frac{N'}{N^\circ}\right)$	$\dot{V}$, cfm: t = 2 days, 0.5 = ε = 0.9		t = 14 days, 0.5 = ε = 0.9	
1 gm	10^4	9.2	6.4×10^4	3.5×10^4	9.1×10^3	5.0×10^3
100 gm	10^6	13.8	9.6×10^4	5.3×10^4	1.4×10^4	7.6×10^3
10,000 gm	10^8	18.4	1.3×10^5	7.1×10^4	1.8×10^4	1.0×10^4

(a) Assuming $N^\circ = 5\ \mu Ci/m^3$ and $V_{TOT} = 10^7$ cu. ft. = $2.8 \times 10^5\ m^3$.

TABLE 4. Analysis of Cyclic Flushing Scenario

Amount Released	$\left(\frac{N'}{N^\circ}\right)^{(a)}$	$\ln\left(\frac{N'}{N^\circ}\right)$	$\dot{V}'$ in cfm (for pumpout time = $t^\circ/2$)(b): t = 2 days, 0.8 = η = 0.99		t = 14 days, 0.8 = η = 0.99	
1 gm	10^4	9.2	8.0×10^4	6.3×10^4	1.1×10^4	9.0×10^3
100 gm	10^6	13.8	1.2×10^5	9.7×10^4	1.7×10^4	1.4×10^4
10,000 gm	10^8	18.4	1.6×10^5	1.4×10^5	2.2×10^4	1.8×10^4

(a) Assuming $N^\circ = 5\ \mu Ci/m^3$ and $V_{TOT} = 10^7$ cu. ft. = $2.8 \times 10^5\ m^3$.

(b) t° = 4 hours.

gas could be stored at 14,000 psi in tanks whose total volume is 10^5 cu. ft., the tanks would then hold 10 room volumes of gas which could be cleaned up over an extended time period (several weeks) by a state-of-the art sized scrubbing system. The cleaned up gas could then be stored for subsequent flushing operations. For the case where the room is evacuated to $1/\eta$ of its normal operating pressure and backfilled every t° hours, one obtains Eq. (3).

$$\frac{dN}{dt} = \frac{\eta}{t^\circ} N \qquad (3)$$

Integrating Eq. (3) for the cases that were considered in Table 3 gives

$$\ln \frac{N'}{N^\circ} = \frac{\eta}{t^\circ} t \qquad (4)$$

The results for this type of atmospheric cleanup are summarized in Table 4. (The ejector/compressor velocities, $\dot{V}'$, are based on pump-out times that are equal to $t^\circ/2$.) In order to keep compressor requirements within reasonable limits, it is necessary that t_0 be on the order of a few hours. A value of t_0 = 4 hours was selected for the study shown in Table 4. (This is considered to be reasonable based on existing experience at several NASA space testing installations.)

Comparison of the results in Tables 3 and 4 shows that there is no practical advantage in terms of gas circulation requirements to the evacuation approach as compared to the more conventional continuous scrubbing approach. The major concern with respect to these massive tritium releases, i.e., soaking of tritium into reactor hardware and building surfaces, will have to be investigated in considerable detail to determine what cleanup durations are acceptable. Clearly, the requirements of cleanup in two days or less imposes large gas circulation requirements and the assocated equipment can be expected to scale accordingly.

A reasonable comprehensive analysis of large-scale atmospheric detritiation for fusion power plants has been presented by Galloway et al.[27] In their study, they consider the question of large-scale cleanup from the standpoint of catalyst performance and cost. Instead of equating processing requirements with removal efficient, ϵ, as was done above, they use first-order kinetic, plug flow reactor design equations to represent catalytic bed performance. This approach permits the evaluation of both performance and cost simultaneously. The method described by Galloway et al.[27] was used in one EPR study[1] to determine air cleanup requirements and costs. The conclusions were that cleanup of a 100 gm tritium release to the atmosphere in a reactor hall having a volume of 10^7 cu. ft. could be made in about two days with a 10^5 cfm air handling system having a capital cost of from 20 to 30 million dollars. Overviews of atmospheric cleanup requirements and prospective strategies for fusion devices are given in references 1, 3, 6, 7, 8, 27, and 29. The status of, and recent progress in, development work on ambient and inert atmosphere cleanup systems is discussed in references 30 through 33.

BLANKET PROCESSING

The capability to recover tritium from breeder blankets at a rate equal to the breeding rate and to maintain a minimal blanket tritium inventory is essential to the concept of DT fueled fusion power plants. Lithium in some form is still regarded as the only substance capable of yielding a breeding gain greater than unity in currently conceived DT fusion reactors. The principal materials options considered to date have been liquid lithium, lithium-containing alloys (e.g., Li-Pb, Li-Al, Li-Si) and ceramics (e.g., Li-Al-O, Li-Be-O, Li-Si-O, Li_2O), and molten salts (e.g., LiF, Li-Be-F). Comparisons[34] of the different tritium breeding concepts have been made with respect to (1) required construction materials, (2) breeding ratio, (3) blanket tritium inventory, (4) prospects for adequate tritium recovery, and (5) ease of containment of tritium within the confines of the blanket structure. These comparisons generally indicate that ceramics and molten salts currently offer the best prospects for meeting anticipated blanket handling and processing requirements and that less development will, in all probability, be required for ceramics and molten salts than for liquid lithium. Fewer facts are available on solid-alloy blankets, but these materials appear to approach the ceramics in terms of inventory and recovery characteristics. Although some encouraging progress has been made in recent years, the development of practical steady-state tritium-recovery techniques for low concentrations remains a major technical uncertainty for liquid-lithium blankets. Also, the magnetohydrodynamic compatibility of liquid lithium with magnetic-confinement concepts (insofar as pumping power requirements and perturbations to plasma confinement are concerned) must still be verified. Nonetheless, in terms of breeding potential, heat transfer characteristics, lithium enrichment, and augmentation of neturon production, liquid lithium still possesses a number of advantages over the other breeder material concepts.

The remainder of this section contains a summary of the status of blanket processing technology for the breeding concepts outlined above. This summary was abstracted from reference 34, and is representative of the opinions delivered at a recent workshop[34] on the subject of fusion reactor blanket technology.

Methods for Processing Liquid-Lithium Blankets

Methods for processing liquid-lithium blankets have received a great deal of attention. The most promising methods currently being considered are: (1) exothermic solid getters (e.g., yttrium and zirconium), (2) permeable metal windows (e.g., niobium- or vanadium-base alloys), (3) liquid-alloy getters formed from rare earth-transition metal eutectics, and (4) molten-salt extraction (e.g., with LiCl-LiF or LiF-LiCl-LiBr). Methods employing cold trapping[37] or batch distillation have, for the most part, been eliminated from further study since neither approach appears capable of achieving tritium inventories in liquid lithium that are near the range of interest (i.e., <10 wppm). Results of studies to determine the thermochemical behavior of solutions of hydrogen isotopes in liquid lithium have been summarized in several recent publications[36,37].

Exothermic getters. Thermodynamic data indicate that solid hydride formers like yttrium and zirconium should be capable of reducing tritium concentrations in liquid lithium to 1 wppm or less. Although limited bench-scale data on extraction of tritium from liquid alkali metals have not been encouraging,[35] further experiments under carefully controlled conditions are recommended. Major technical uncertainties that need near-term exploration are potential passifying effects of impurities (including principally the nonmetallic elements O, N, and C) and methods for regeneration of the getter after loading with tritium. If these studies are sufficiently encouraging, subsequent investigations would have to examine (1) the getter solubility in lithium and the potential for entrainment of degraded getter in the lithium returning to the reactor; (2) the effects on neutronics, corrosion, and mass transport; and (3) the effects of getter composition and morphological characteristics on extraction and recycle efficiency.

Permeable windows. Although the permeability of hydrogen isotopes through most metals normally occurs at a rate that is more of a nuisance than anything else, highly permeable metals (e.g., niobium or vanadium) at elevated temperatues may be used as window materials through which tritium could be extracted from liquid lithium. Calculations using existing permeation data (collected at hydrogen pressures many times higher than those anticipated in fusion reactor blankets) generally indicate that tritium concentrations in lithium in the range from 1 to 10 wppm can be maintained with large but reasonable window areas. Major technical uncertainties in need of near-term investigation relate to (1) "fogging" effects of impurities (e.g., O, N, and C) and mass-transported metals (in dissimilar metal systems only) that may be deposited on the upstream side of the window, (2) "fogging" effects on the downstream side of the window, and (3) fundamental limitations associated with surface kinetics. The possible use of downstream recovery methods employing either liquid getters or protective coatings coupled with gaseous getters should eventually come under study. Limitations imposed by temperature effects on window integrity will eventually need study as well.

Liquid alloy getters. Recent work at LASL[39] indicates that binary liquid eutectics consisting of a rare-earth metal (e.g., Ce, La, Y) and a transition metal (Co, Ni, Fe, or Mn) make highly effective getters for removing hydrogen isotopes from liquid lithium. In principle, these liquid alloys would be capable of maintaining tritium concentrations in lithium blankets well below 1 wppm. Current uncertainties in this technique include mutual solubilities of the getter alloy and lithium and the difficulty in recovering tritium from the getter alloy. Compatibility of the getters with stainless steel needs to be investigated. Because of the potential promise of these liquid getters, near-term efforts should include examination of the above-mentioned problems. Identification of other low melting compositions (possibly ternary and higher-order mixtures) is recommended.

Molten-salt extraction. Work currently under way at ANL[37,38] indicates that molten-salt extraction may be a suitable means of recovering tritium from liquid lithium. Results to date show that adequate distribution coefficients (2 to 4 on a volumetric basis in favor of the salt) can be achieved, and potentially suitable methods for recovery of tritium from the salt are being investigated. This technique may permit the maintenance of tritium levels as low as 1 wppm in liquid lithium. Areas requiring study in the near future include (1) effects of mutual solubilities on both neutronics and salt processing, (2) compatibility of materials, (3) survey of suitable minimum-entrainment contacting and separating methods, and (4) demonstration of suitable means for recovering tritium from the salt.[38]

Tritium Recovery from Solid Breeder Blankets

Analytical[40,45] and experimental[40] work on solid blankets has been restricted mainly to performance characteristics that might influence tritium recovery, inventory, and containment. Results achieved in the BNL program[40] on ceramic breeding materials have been reasonably encouraging with respect to steady-state recovery under minimum inventory conditions (<<1 wppm). Similar results for solid Li-alloys are less understood, but efficient recovery at low inventory is indicated. Important near-term exploration of ceramic materials should include (1) determination of the effects of irradiation to high burnups on dimensional stability, tritium release rates, and chemical stability toward the sweep gas; (2) investigation of the dependence of performance characteristics on morphology (particle size, size distribution, pore structure, etc.); and (3) compatibility with containment materials and with other companion substances, including moderators (B and C) and neutron multipliers (Be and Pb). Effects of chemisorption, implantation, or chemical fixation of tritium on interior blanket structures should be examined in the context of the "hands-on" maintenance expectations for minimum-activation designs. Eventually, in-reactor testing of candidate solid breeder materials under realistic conditions (including simultaneous breeding and tritium extraction) should be made. Design optimization of sweep gas processing methods must ultimately be completed.

Tritium Recovery from Molten-Salt Blankets

Recovery of tritium from molten-salt blankets appears to be reasonably straightforward,[42-44] and inventories well below 1 wppm should be achieveable. Past analyses of sparged and unsparged desorbers have indicated that efficient removal of tritium from the salt is readily attainable in forms well suited for efficient regeneration. These analyses require experimental demonstration under conditions which take proper cognizance and control of (1) the oxidation potential of the salt system, (2) interactions between the salt and bounding walls, and (3) the influence of magnetic fields in promoting localized electrolysis.

UNIQUE PROBLEMS

In addition to the fuel circulation and processing hardware and the various containment devices, there are a number of other special systems that may be essential to the operation of at least near-term DT reactors and whose presence could lead to significant interfacing problems with the tritium handling systems. Two such systems are neutral beam injectors and first wall cooling circuits. Some considerations related to these special interfacing problems are summarized below. As other essential reactor systems that have a direct interaction with, or that have access to, the tritium systems are identified, they too must be carefully characterized in terms of the magnitude and design impact of interfacing requirements.

Neutral Beam Injector Interfacing

Recent conceptual design studies have shown that experimental tokamak and mirror reactors may require energetic particle insertion to reach ignition. As currently conceived,[52,53] the neutral injector systems that would be used to provide these energetic particles represent a direct access to the plasma chamber, and, hence, to the tritium contained in it during a burn cycle. The beam line, the neutralizer, and the accelerator itself are all subject to tritium backflow from the plasma chamber. Thus, tritium can enter the neutral beam pumping systems and the neutralizer vapor circulation system, and pass through the electrostatic grid structure into the grid coolant. The objective of neutral beam injector interface studies should be to identify significant tritium interactions and to evolve and test designs that will permit the making of a workable injector/reactor interface. This effort should include the determination of tritium containment criteria for neutral injectors and the development of a maintenance methodology for tritium contaminated injector systems.

Tritium Migration to the First-Wall Cooling Water

The migration of tritium in and through the thermally hot structures surrounding the plasma of a DT burning reactor should be the subject of a whole paper in itself, since this migration will probably turn out to be the major contributor to tritium losses from any fusion power plant. In this section, however, the only case that will be addressed will be one in which the first wall of a near-term experimental reactor is cooled with a fluid that neither contains breeding material nor interfaces in any way with a breeder blanket; i.e., the only source of tritium entry into the coolant fluid is by permeation of fuel from the plasma chamber through the first wall structure and into the coolant channels. Although a variety of coolant fluids and structural materials could be considered in the context of this discussion, only helium and water contained in stainless steel first wall assemblies will be addressed here for purposes of illustration.

The selection of water as a coolant for the first wall blanket and shield of any fusion device raises concern regarding the consequences of tritium migration into and through the cooling circuits. The problem of tritium permeation through austenitic construction materials has long been recognized as a major area of concern for fusion power plants. Although the principal focus of attention to this problem has been on migration of blanket tritium through heat transfer circuits and eventually to the environment, the tritium in the plasma chamber is also subject to migration; hence, the absence of a breeder blanket would not necessarily obviate concern for tritium releases resulting from the heat transfer circuitry. For purposes of analysis, permeation was assumed to occur across the entire surface of a first wall having a thickness of 5 mm and an area of 6 x 10^6 cm^2 (approximating presently conceived tokamak EPR's[1]). The total volume of the first wall cooling water circuit was assumed to be 10^5 liters and the tritium partial pressure in the plasma chamber was estimated at 10^{-4} Torr. Existing hydrogen permeation data for Type 316-SS (data from reference 23 were used in this case) were divided by $\sqrt{3}$ to correct for H/T isotope effects. Results of calculations made under these conditions are summarized in Fig. 1 together with related data for existing heavy water reactors. The concentrations of tritium in the first wall cooling water circuit are found to be in the range of those reached in the primary (D_2O) circuit of typical experimental heavy water reactors[46-48] after a period of about two years. It is, therefore, reasonable to assume that the inventory of tritium in the cooling water for the above case could be managed using practices employed in existing heavy water reactors. The drainage, consolidation, and disposal of the entire first wall cooling water inventory (10^5 liters) every two to three years should pose no insurmountable problems, and is probably preferable (from an economic viewpoint) to installing a tritium extraction plant[46] for the first wall coolant.

In the case of pressurized helium cooling, the assumption is usually made that any tritium entering the coolant stream would be rapidly converted to HTO through reaction with the low levels of H_2O and O_2 impurity expected to be present in helium circuits. This HTO could then be gettered or desiccated out of the helium in a relatively straightforward manner, such that tritium buildup beyond trace levels is not possible. For the cases considered in Fig. 1, but with helium cooling, the total tritium handled in the helium scrubbing system would not exceed 10 grams/year.

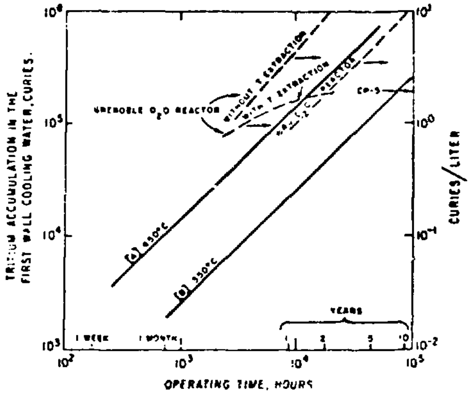

FIGURE 1. Tritium Accumulation in the First Wall Cooling Water Circuit. Material = 316-SS, Wall Area = 6 x 10^6 cm^2, Wall Thickness = 5 mm, Coolant Volume = 10^5 liters, Tritium (T_2) Pressure = 10^{-4} Torr.

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