**PORTIONS
OF THIS
DOCUMENT
ARE
ILLEGIBLE**

TITLE: THE TRITIUM SYSTEMS TFST ASSEMBLY AT THE LOS ALAMOS
SCIENTIFIC LABORATORY

AUTHOR(S): J. L. Anderson

SUBMITTED TO: 3rd American Nuclear Society Topical Meeting
on the Technology of Controlled Nuclear Fusion,
Santa Fe, NM, May 9-11, 1978

NOTICE **IN ONLY**

**PORTIONS OF THIS REPORT ARE ILLEGIBLE. It
has been reproduced from the best available
copy to permit the broadest possible avail-
ability.**

By acceptance of this article, the publisher re-
cognizes that the U.S. Government retains a non-
exclusive, royalty-free license to publish or repro-
duce the published form of this contribution, or to
allow others to do so, for U.S. Government
purposes.

The Los Alamos Scientific Laboratory requests that
the publisher identify this article as work performed
under the auspices of the Department of Energy.

MASTER

los alamos
scientific laboratory
of the University of California
LOS ALAMOS, NEW MEXICO 87545

An Affirmative Action/Equal Opportunity Employer

NOTICE
This report was prepared in connection with work
performed by the Los Alamos Scientific Laboratory for
the United States of America. The publication of
this report is authorized by the Director of the
Los Alamos Scientific Laboratory. The copyright in
this report is reserved by the Los Alamos Scientific
Laboratory. The Los Alamos Scientific Laboratory
assumes no responsibility for the use or misuse of
the information contained herein. The Los Alamos
Scientific Laboratory does not warrant the accuracy
or completeness of the information contained herein.
The Los Alamos Scientific Laboratory does not
assume any liability for the use or misuse of the
information contained herein.

THE TRITIUM SYSTEMS TEST ASSEMBLY AT THE
LOS ALAMOS SCIENTIFIC LABORATORY

J. L. Anderson

LOS ALAMOS SCIENTIFIC LABORATORY, UNIVERSITY OF CALIFORNIA
LOS ALAMOS, NEW MEXICO 87545

The Tritium Systems Test Assembly (TSTA)⁽¹⁾ is dedicated to the development, demonstration, and interfacing of technologies related to the deuterium-tritium fuel cycle for fusion reactor systems. The principal objectives for TSTA are: (a) demonstrate the fuel cycle for fusion reactor systems; (b) develop test and qualify equipment for tritium service in the fusion program; (c) develop and test environmental and personnel protective systems; (d) evaluate long-term reliability of components; (e) demonstrate long-term safe handling of tritium with no major releases or incidents; and (f) investigate and evaluate the response of the fuel cycle and environmental packages to normal, off-normal, and emergency situations.

PROJECT SCOPE

TSTA Function

The TSTA is dedicated to the development, demonstration, and interfacing of technologies related to the deuterium-tritium fuel cycle for fusion reactor systems. The first such system to be built will be a Tokamak TNS (The Next Step) or ITR (Ignition Test Reactor) deuterium-tritium burning machine. This TNS will be followed by a TNS-upgrade or an Experimental Power Reactor (EPR). Either of these later devices will build and expand on the tritium handling system used in TNS. The objectives of the TSTA Project are to design, fabricate, construct, test and bring into operation the Tritium Systems Test Assembly at LASL within approved envelopes of technical schedules and cost baselines. The principal program objectives for TSTA can be stated as:

- (1) demonstrate the fuel cycle for fusion power reactors;
- (2) develop, and test environmental and personnel protective systems;
- (3) develop, test, and qualify equipment for tritium service in the fusion program;
- (4) provide a final facility that can be used

for demonstration and as an example that could be directly copied for a D-T burning fusion machine;

- (5) demonstrate long-term reliability of components;
- (6) demonstrate long-term safe handling of tritium with no major releases or incidents; and
- (7) investigate and evaluate the response of the fuel cycle and environmental packages to normal, off-normal, and emergency situations.

The TSTA will consist of a large gas loop, Fig. 1, which can simulate the proposed fuel cycle for a fusion facility. The loop, as shown, does not include any specific fuel injection system, but will be sufficiently versatile so that such systems can be added as the design requirements are better defined. The gas loop will be designed to handle up to 360 gram moles per day DT. This flow will provide cycle operating experience on a scale that is equal to or greater than the full-scale fuel cycles currently being addressed for TNS and EPR systems.

To accomplish this at TSTA will require an on-site tritium inventory of approximately 200 g.

*This work supported by the Office of Fusion Energy
U.S. Department of Energy.

FIGURE 1. TSTA - Main process loop and auxiliary systems.

The assembly to do this will consist of a number of integrated subsystems as follows:

- Vacuum pumping (VAC)
- Transfer pumping (TFU)
- Fuel cleanup (ash and impurity removal) (FCU)
- Hydrogen isotope separation (ISS)
- Fuel mixing and injection (including impurity simulation)
- Interfaces with external systems (neutral beam, coolant breeding blanket systems)
- Fuel storage

Essential functions associated with the overall assembly include:

- Tritium waste treatment (TWT)
- Emergency room cleanup and effluent controls (ETC)
- Waste disposal (SWD)
- Tritium monitoring (TM)
- Personnel safety
- System maintenance
- Instrumentation
- Computer control and simulation
- Analytical system (i.e. gas analysis), and
- Quality assurance procedures

Project Interface with the U.S. Fusion Program

The development of TSTA at this time will permit design and evaluation of the required fuel cycle well in advance of the final design of a TNS tritium system. This timely accumulation of data will

insure that the design and development of the tritium system will not be a pacing item in the design of TNS. The development of the tritium technology required at TSTA for the fuel cycle and environmental packages will not require any large scientific or technical breakthroughs. There currently exists within DOE and specifically within the Los Alamos Scientific Laboratory (LASL) a considerable amount of experience and expertise in tritium technology. This experience and expertise will be used at TSTA to insure that the design of the facility will reflect the most current information and philosophy pertaining to tritium technology.

While the design of any specific subsystem may not require a significant technology advancement, the integration of all of these subsystems into one package is a significant advance. There currently exists nowhere within the Department of Energy complex a facility which integrates all of these subsystems into a continuously operating, computer controlled system. The flow rate and the total quantity of tritium to be handled in a fusion reactor on a routine basis necessitate the establishment of TSTA as a demonstration facility. Table 1 gives a comparison of the projected tritium inventories and release goals for five fusion facilities. This comparison serves to emphasize the need for the timely development of TSTA. The Tokamak Fusion Reactor, TFTR, is currently in the early stages of construction at Princeton, NJ. All of the other projected facilities in the Table will follow TSTA and will utilize the information obtained at TSTA.

TABLE 1. Tritium Inventories in Fusion Facilities

Facility	T ₂ Inventory (kg)	Fuel Feed (kg/day)	T ₂ Release Goal (Ci/day)
TFTR	0.005	0.001	1
TSTA	0.2	0.3-1.5	1
TNS	0.25	0.25-0.5	1-10
EPR	4-6	1-2	3
^a DEMO	6-12	2-6	3-10

^aDEMO = 1000-MWe net output

A very significant goal at TSTA will be to demonstrate that these large quantities of tritium can be handled safely on a routine basis. The establishment of environmental and safety packages for tritium systems within the Office of Fusion Energy (EHE) program, will be a significant milestone. The TSTA will serve as a baseline facility that will provide a large data base that can be used to establish future guidelines and requirements for fusion facilities.

The TSTA project has a baseline Total Estimated Cost (TEC) of \$13.2 M of this, \$4.8 M is for LSL operations, \$6.9 M for hardware and equipment procurement and installation and \$1.5 M is contingency funding. The baseline schedule projection call for TSTA to be on-line and operating by mid FY-1981.

TSTA Overview

TSTA can be broken into four major subdivisions:

- (1) the main process system,
- (2) the environmental and safety systems,
- (3) supporting systems, and
- (4) the physical plant.

The main process system is designed to provide a demonstration of all of the components of a DT fuel cycle. A somewhat more detailed block diagram of the process system is shown in Fig. 2. Deuterium and tritium will be mixed and injected into a representative vacuum chamber together with impurities expected to accumulate in a real fusion burn. Helium

will also be injected to represent the "ash" from such a burn. The vacuum system is arranged so that it may be operated either in a closed loop by itself, or completely integrated with the remainder of the process system. This system will be capable of testing the operation of cryopumps having up to 100-cm-diameter port. Pumps with pumping speeds of $10\text{--}50\text{ m}^3/\text{sec}$ will be evaluated and tested at TSTA.

A torus mock-up assembly will simulate the volume of one-fourth of one section of a TNS toroidal vacuum chamber. Vacuum pumps will be connected to the chamber by one or more ducts of up to 90 cm diameter. The premixed plasma exhaust gases will be injected into the chamber during a simulated plasma burn cycle. During the evacuation cycle, these gases will be pumped through a fast-acting valve at the exhaust port. Heating of the simulated exhaust gases will be provided by heating the high-density fuel in the small injector volume. This heat will be removed by bends in the exhaust duct and a refrigerated baffle ahead of the pumping system. Molecular flow calculations on current TNS designs indicate that no special provisions need to be made for post-burn cooling.

The initial torus evacuation system will be comprised of a one half-scale, compound, DT-cryocondensation/He-cryosorption pump and means for its regeneration. Regeneration will be accomplished by using a combination of oil-free metal-membrane pumps and mechanical boosters. The VAC cryopump will incorporate separate temperature control of its two cryogenic panels, so schemes for separating the helium "ash" from the DT "fuel" during regeneration can be investigated. Initial and post-maintenance evacuations will be provided by an intermediate-size oil-sealed mechanical pump and one mechanical booster operating in series. The scenarios for initial evacuation, postburn torus evacuation, helium and DT recovery from cryopanel, and pre-maintenance evacuations will be traced by referring to Fig. 3.

A. Initial Evacuation.

The normal roughout will be through path A-F-J-K-M. A conventional oil-sealed mechanical pump

FIGURE 2. Flow diagram - Tritium Systems Test Assembly.

FIGURE 3. VAC schematic.

in path K will provide adequate pumping speed to reduce the torus pressure to 2 kPa within 1 h. At 2 kPa the mechanical booster can be started, and torus pressure will drop to 25 Pa in another few min. During the initial roughout, the main cryopump may either be on refrigerated standby, or on its own roughout/bakeout cycle. In the latter case torus roughout will be by path B-C instead of A. In either case, initial evacuation below the 25 Pa level may be effected with the air-bearing turbomolecular pump via B-D-L-M. During all phases of initial torus pumpdown, all pumps exhaust through a tritium-monitored duct to the stack or to the TWT, depending on the tritium concentration. Normal discharge in this mode will be directly to the stack, but all failure modes or a tritium alarm will automatically route all pump discharges through the TWT. Appropriate traps will be distributed through the system to collect condensables (at B) or to inhibit migration of oil vapor to high vacuum sections of VAC (F).

B. Postburn Torus Evacuation.

This is the primary function of VAC. With both

panels of the cryopump refrigerated to their operating temperatures, evacuation will take place through the high-conductance path B. A 76 K baffle will remove condensables and reduce the temperature of the postburn torus gas before it enters the cryopump. Heating of the gas to the simulated postburn temperature will be done before injecting into the torus volume.

C. Cryopanel Regeneration.

Recovery of the saturation gas load from the cryopanel will occur in two stages. If it is possible to warm the inner helium-cryosorption panel without disturbing the DT frozen on the outer chevrons, this helium can be removed by a gas-bearing turbopump via path L-M. The normal discharge of the turbopump, which will consist of helium, nitrogen from the pump bearings, and small quantities of DT, will be routed through the TWT. A flow-through uranium bed or sorption trap ahead of the turbopump can be used to recover any excess DT before it is mixed with the bearing gas. After most of the helium has been removed the turbopump can be valved off, and recovery of the DT can commence via path C-F-G-H, using a series of metal-bellows and mechanical booster pumps. The pumping speed of this system should be adequate to maintain peak pressure below 100 Pa and to reduce it ultimately to less than 1 Pa. The cryopanel can be warmed to the shield temperature (77 K) during this stage, as a significant inventory of liquid helium is not required inside the cryopump. Thus recovery of the DT can be accomplished rapidly at fairly high pressures. If no separation of He and DT is attempted during panel regeneration, the two panels will be warmed concurrently and the mixed gases removed through path C-F-G-H. Either way the gas regenerated from the cryopump is stored in an accumulator awaiting transfer either to the FCU in the full operational mode, or reinjection into the torus in the VAC closed-loop mode. Condensables trapped on the path B cryotrap will be removed occasionally by the turbopump via path E-L-M, discharging to TWT.

D. Pre-maintenance Evacuation.

Before any maintenance can be performed on the VAC, all tritium must be safely stored and the lines

associated with the faulty component must be evacuated. Most of the DT gas can be stored in the accumulator (path R), and smaller amounts remaining in the lines can then be absorbed onto the sorption pump (path P-Q) assisted by the mechanical boosters. Provisions will be made to back-fill the torus mock-up volume with argon, and any connecting components can be flushed with argon from this source. Before lines are breached and components are repaired or replaced, the mechanical vacuum pump (K) will provide the final evacuation after appropriate argon backflushing.

Cryogenic distillation has been chosen as the method utilized to separate the hydrogen isotopes. To avoid persistent plugging of the distillation columns it is necessary to reduce condensable impurities to about 1 ppm. The fuel cleanup system to achieve these purity levels is the most complicated portion of the process system because it is difficult to specify either impurity components or composition levels at this time and because of the ultimate purification required.

The fuel cleanup system currently under consideration for TSTA is based on the selective adsorption of impurities on molecular sieve and activated charcoal traps. Experimental studies are currently underway to study several alternate schemes for fuel cleanup. Figure 4 shows the current preferred scheme for the fuel cleanup system.

The cryogenic distillation, shown in Fig. 5 will be accomplished by means of four distillation columns. Four product streams are: (1) pure DT, (2) pure T_2 , (3) pure D_2 , and (4) a waste stream of mostly HD.

The DT + T_2 will be mixed and injected into the vacuum chamber as fuel. The D_2 stream will be

FIGURE 4. Fuel Cleanup System.

THE ISS IS DESIGNED TO PROVIDE PURE D_2 FOR NEUTRAL BEAMS, HD WASTE FREE OF T_2 , AS WELL AS STREAMS FOR REACTOR REFUELING AND MATERIALS STUDIES

FIGURE 5. Isotope Separation System

injected separately thus simulating neutral beam injectors, including fuel back-streaming through pumping systems.

The isotope separation system is discussed in detail by Bartlit, Sherman and Denton in a paper presented at this meeting. (2)

The environmental and safety systems consist of five major subsystems:

- (1) Tritium Waste Treatment (TWT),
- (2) Emergency Tritium Cleanup (ETC),
- (3) Experimental Contamination Studies (XCS),
- (4) Tritium Monitoring (TM), and
- (5) Secondary Containment (SEC)

The TWT system is designed to provide tritium removal from all routinely generated gaseous effluents prior to the release of these effluents to the atmosphere. The TWT will also provide for detritiation of the environmental gas in all secondary containment volumes. This system is very similar to a working cleanup system described by Andersen et al. (3)

In the event of an accidental spill of T_2 into the building, the emergency tritium cleanup system, dedicated to the treatment of room air, is called into use. It will operate on a similar principle to the TWT, but with a flow rate of 1000 cfm. The basic operation of the ETC is shown in Fig. 6. The ETC is described in a paper by Muller (4) at

The experimental contamination studies will be set up in a separate room at TSTA. This portion of TSTA is devoted to deliberate high-level contamination of components and structural materials and a study of cleanup methods, kinetics, etc.

'Secondary containment will be employed wherever a hazardous quantity of tritium is at risk based on the structural design of the particular subsystem and a risk evaluation.

It is designed for redundancy to avoid emergency shutdown in the event of component failure. The MDAC will permit examination of off-normal and emergency operating conditions which will be deliberately demonstrated. It will be equipped with both an uninterruptable power supply and an emergency generator set.

1. J. L. Anderson and R. H. Sherman, "Tritium Systems Test Assembly Design for Major Device Fabrication Review," Los Alamos Scientific Laboratory Report LA-6855-P (June 1977).
2. J. R. Bartlit, R. H. Sherman, and W. H. Denton, "Hydrogen Isotope Distillation for the Tritium Systems Test Assembly," Proc. of 3rd ANS Topical Meeting in the Technology of Controlled Nuclear Fusion, Santa Fe, New Mexico, (May 1978).
3. J. L. Anderson, F. A. Damiano, and J. E. Nasise, "Tritium Handling Facilities at the Los Alamos Scientific Laboratory," Proc. of 23rd Conference on Remote Systems Technology, 77 (1975).
4. M. E. Muller, "Conceptual Design of an Emergency Tritium Clean-Up System," Proc. of 3rd ANS Topical Meeting on the Technology of Controlled Nuclear Fusion, Santa Fe, New Mexico, (May 1978).
5. M. E. Muller, "Conceptual Design of an Emergency Tritium Clean-Up System," Los Alamos Scientific Laboratory Report LA-7136-MS, (January 1978).