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LITHIUM TEST MODULE ON ITER:
ENGINEERING DESIGN OF THE TRITIUM RECOVERY SYSTEM

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ABSTRACT

The design presented is an overview of the tritium recovery system for a lithium module on an ITER type reactor. The design of a tritium recovery system for larger blanket units, sectors, etc. could use the information developed in this report. A goal of this design was to ensure that a reliable, integrated performance of the tritium recovery system could be demonstrated. An equally important goal was to measure and account for the tritium in the liquid lithium blanket module and its recovery system in order to validate the operation of the blanket module.

INTRODUCTION

The ITER (International Tokamak Engineering Reactor) will provide a fusion environment for engineering tests of different nuclear systems for use in fusion power reactors. A primary nuclear system to be tested is the breeder blanket and its associated tritium recovery system. As a part of the ITER design, the engineering design of different blanket modules was done.(1) This report summarizes the design of the tritium recovery system for the liquid lithium blanket, a prime candidate for a fusion power reactor. A goal of this design was to ensure that a reliable, integrated performance of the tritium recovery system could be demonstrated. An equally important goal was to measure and account for the tritium in the liquid lithium blanket module and its recovery system in order to validate the operation of the blanket module.

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TRITIUM RECOVERY SYSTEM

The reference tritium recovery system for the lithium breeder module on ITER/Tiber, is a molten salt extraction system. The main components of a molten salt extraction system are a centrifugal contactor unit, an electrolysis unit, and a getter system. The extraction system is linked to several secondary tritium systems to minimize tritium losses from the lithium module, the module's heat exchanger and other tritium subsystems. The design of each of these systems is described below.

Molten Salt Recovery

The molten salt method (2,3) of recovering tritium from liquid lithium is a three-step process. First, lithium which contains tritium is intimately mixed with a salt mixture of lithium halides in a centrifugal contactor. The tritium which is in the form LiT, lithium tritide, preferentially distributes to the salt phase. The salt phase, which is denser than the lithium, is separated from the lithium by centrifugal force. Second, the salt phase is circulated to an electrolysis unit in which the LiT is dissociated to form molecular tritium. The molecular tritium is swept from the salt phase by bubbling helium through the electrolysis unit. Third, the helium is circulated to a cleanup unit which removes the tritium and impurities from the helium gas.

Although individual steps in the molten salt recovery method have been demonstrated, the integrated process has not been demonstrated. The engineering problems associated with the integrated process are unknown. Prior to a detailed design of a system for a commercial fusion reactor, engineering prototypes for the centrifugal contactor, the electrolysis unit and the cleanup unit would be designed, built and tested. The testing would be conducted both in independent and integrated operation to assess the actual magnitude of the problems. A chemical engineering model would be available to predict behavior of the molten salt unit as a function of system size, throughput and impurities.

We have used experimental information (4) for a room temperature aqueous/organic system to size the centrifugal contactor for the lithium module. The densities of this aqueous/organic system approximate those of lithium and molten salt. Figures 1 and 2 show the contactor throughput as a function of rotor diameter and the contactor volume as a function of rotor diameter. The needed throughput to attain 1 wppm tritium in the 2000 L of lithium in a lithium module is 300 L/hr. We have assumed a distribution coefficient of 0.5, a contactor efficiency of 0.9 and an electrical efficiency of 0.85. Since the lithium to molten salt volume should be 1:3 to ensure good separation of the lithium from the salt, the total throughput per hour is 1200 L/hr. Using Fig. 1, a 13 cm diameter contactor is needed to handle this throughput. Since as a rule of thumb, the length to diameter ratio for a contactor is 1.6, the height of the contactor is 21 cm.

Entrainment of either the lithium or the molten salt in each other is undesirable, therefore, the centrifugal contactor is not a stand-alone unit but three contactor units in series. Three contactors in series ensure that lithium is not entrained into the molten salt and that molten salt is not entrained into the lithium during operation. Separation of the lithium from the salt prevents recombination of the dissociated tritium with excess lithium in the electrolyis unit. Separation of the halide salt from the lithium keeps the halides out of the fusion reactor in which they could be activated, could enhance corrosion, and could precipitate out in the colder parts of the lithium coolant loop. Although the three contactors in series prevent entrainment, mutual solubility of the two liquids can cause the above problems to a limited extent. The upper limit is assumed to be the solubility limits of the two liquids. A duplicate set of contactors is provided to increase system reliability. If a problem arises, the lithium flow is switched to the alternate set.

When we designed the centrifugal contactor, we considered the effect of the magnetic field on the circulating lithium. The lithium was assumed to rotate as a rigid body. Therefore, we calculated the power needed to overcome the torque generated by this rotation in a magnetic field. Unless the contactors are located in an area in which the reactor's magnetic field strength is at the earth's normal field, shielding is needed which complicates their design. The magnetic field at a distance <5 m from a magnetic fusion reactor is of the order of 500 gauss (5). (The tritium recovery system is <5 m from the reactor to minimize the total volume of lithium at risk.) Lithium, which is rotating at 2000 rpm (the expected rpm to achieve tritium separation) in a contactor with dimensions of 13 cm diameter and 21 cm height, would require 2 kwatt of power to counteract the magnetic forces generated. This power requirement is equal to the power needed for the unit's operation. With three centrifugal contactors on-line, the power needed to counter the magnetic force is 6 kwatt. Since the contactor may not be located perpendicular to the reactor's field, the magnetic force may not be evenly distributed. Therefore, each contactor requires magnetic shielding which is at least 0.6 cm thick and extends 3 cm beyond the ends of each contactor unit. To minimize total volume, each contactor has its own magnetic shielding. For larger units, additional power is required to overcome the magnetic forces, i.e., a 25 cm diameter unit needs 26 kwatts.

For the electrolysis unit, we chose helium as the sparge gas to minimize the generation of gamma gaseous species, i.e., Ar-41, in gas entrained to the fusion reactor. Each electrolysis unit has an outer jacket to control tritium permeation from units at >500°C. We were concerned with the lifetime of the electrodes in the electrolysis unit. Since impurities from metal corrosion (metal oxides or lithium oxide) may reduce the lifetime of the electrodes or may increase the overvoltage required by clogging the electrode, we provided duplicate units. Replacement is then easily effected; in addition, excess capacity is available if required. To minimize metal oxide impurities, we used a cold trap to purify the salt. We also had a halogen sparge to minimize the amount of these species in the gas stream.

The cleanup unit for the gas stream separates the helium sparge from the tritium and impurity species, i.e., radioactive gases and entrained halogens. A sacrificial bed is used to trap the entrained halogens. A palladium diffuser or a hydrogen getter after the sacrificial bed separates the tritium from the helium and radioactive gases. We tried to reduce tritium residence times by having duplicate getter systems and switching between them frequently.

Lithium Module

The lithium module has three sections (1). The front section near the plasma consists of a vanadium structure and liquid lithium coolant. The middle section consists of a vanadium structure, liquid lithium coolant and calcium oxide or tungsten as a neutron reflector. The back section consists of a stainless steel structure and helium coolant. The first two sections operate at 500°C. The back section at <200°C. The module has the overall dimensions of 2m x 2m x 2m. Around the lithium module is a stainless steel structural envelope which contains helium gas. The helium is monitored to measure tritium permeating from the lithium module. The tritium is removed by passing the helium through a getter.

Heat Exchanger

To minimize the danger of lithium and water interactions, the heat exchanger for the lithium module is a three stage unit. In the first stage, the working fluids are lithium and NaK, in the second stage, NaK and an organic, and in the third stage, an organic fluid and water. The first stage is a shell-and-tube heat exchanger with lithium in double-wall tubes of vanadium bound to stainless steel. The tubes are 1 mm thick vanadium next to the lithium and 1 mm thick stainless steel next to the NaK. The heat transfer area is 20 m^2. A cold trap on the first stage of the heat exchanger collects the tritium which permeates across the double-wall tubes.

Lines

The lines from the lithium module to the tritium recovery system and to the heat exchanger are also double-wall tubes of vanadium bound to steel with stainless steel on the outside. Helium between the two metals is periodically monitored to measure tritium losses.

Interface to ITER Plasma Tritium System

The tritium recovered from the liquid lithium breeder module is passed via an interface (a stainless steel tank) to the plasma processing system on ITER/Tiber. The interface decouples the operation of the tritium recovery system for the lithium module from that of the plasma processing system. This minimizes the probability of a failure in the lithium system from propagating to the main tritium system. It also minimizes contamination among the different blanket tritium recovery units since the stainless steel tank serves as a general interface for all blanket module tritium recovery systems.

Instrumentation

Tritium tests in ITER/Tiber for the liquid lithium breeder module and its tritium recovery system will be used to confirm the predictive capability of engineering models, to demonstrate integrated performance and reliability of the tritium recovery system for the lithium module in a fusion environment, and demonstrate the limits on the tritium mass balance for a lithium blanket and its tritium recovery system. Multiple tests on-line and off-line will be required during the six months of full-power operation allotted to the lithium module and its tritium recovery system in the ITER/Tiber operating schedule. The parameters measured include: the total tritium recovered from the tritium recovery system; the total tritium recovered from secondary systems (these include the intermediate heat exchanger, the lithium module's envelope system, irradiated structural materials, etc.); tritium losses in the recovery system; the processing efficiency of the centrifugal contactor as a function of time; the processing efficiency of the electrolysis unit as a function of time; the impurities in the lithium; the impurities in the molten salt; and the post-irradiation tritium content in all materials. Instrumentation is provided for each of these needs.

The reactor's magnetic field affects the type of instrumentation chosen. Sensitive instruments may be affected by as little as 2 gauss, therefore they will require individual magnetic shields. For data transmission either fiber optics or telemetry will be required.

Schematic

A schematic of the tritium system design for the liquid lithium breeder module is shown in Fig. 3. The tritium systems include the molten salt recovery system, a cold trap system for the main heat exchanger, a getter system for the cover gas for the lithium module and a simple interface to the plasma processing system. Duplicates of each of these units are shaded. The duplicates provide increased reliability and increased capacity. The minimum volume needed for the tritium processing equipment is 130 m^3 if access space is included, 40 m^3 being space for the heat exchangers. The equipment in this space includes dump tanks for the lithium and the salt, pumps for the lithium and the salt, the contactors, electrolysis units, and getter units for the molten salt extraction system, the cold traps for NaK and the getters for the helium cover gas for the lithium module. We estimate that the capital cost for all components of this tritium recovery system for the lithium module is $5 M.

TRITIUM INVENTORY

We need to know the amount of tritium needed to saturate all components of the breeder blanket and its tritium recovery system to predict the time needed to attain steady-state processing information. To provide an estimate, we calculated the tritium inventory and the tritium permeation losses associated with each tritium subsystem. These are summarized in Table 1. The total inventory in the systems is ~2 g whereas, the total tritium production in the lithium module is 0.5 to 1.0 g/24-hour full-power day. Therefore, the minimum time before steady-state conditions are achieved is 1-4 full-power days. The actual time required to achieve steady-state will be greater since the reactor will not be operated on a 24-hour schedule. In addition, slow tritium diffusion rates may slow-down attainment of equilibrium conditions.

Table 1. Lithium Blanket: Unit Dimensions, Estimated Tritium Inventory, and Tritium Permeation Rates at Steady State Conditions (500 ^{o}C, 10^{-8} Pa)

Location	Inventory (g)	Permeation (Ci/d)	Volume (m^3)	Area (m^2)	Thickness (m)
Tritium Recovery					
-Molten Salt	0.02	<10-5	0.5	--	--
-V Contactor	<10-4	0.19	1.7(10^{-3})	0.27	0.006
-Getter	<0.1				
Lithium Module					
-Lithium	0.5	--	2	--	--
-Vanadium	0.3	0.6 (Plasma)	0.08	1.24	0.01
		44 (CaO/W)	--	17.6	0.002
-CaO (1 wppm)	<0.8	--	0.26	--	--
-Tungsten	<10-4	--	--	--	--
-Getter (He)	<10-4				
Heat Exchanger					
-Vanadium	0.08	116	0.02	23	0.001
-Stainless Steel(1)	0.08	0.02	0.02	23	0.001
-NaK(100C Trap)	<0.01	--	0.75	--	--
Vanadium Lines	0.1	<10^{-3}	0.03	5.5	0.005
Total	2				

1) Maximum Temperature is 300°C.

CONCLUSIONS

The following goals drove the design of the tritium recovery system: 1) minimize the tritium inventory in the lithium breeder module, the tritium recovery system, and other associated systems; 2) minimize permeation losses in the intermediate heat exchanger, lithium module, etc.; 3) prevent cross-contamination problems in linking this recovery system to the main fuel processing system; 4) minimize magnetic effects for the centrifugal contactor; 5) minimize instrumentation problems due to the magnetic field; and 6) provide maximum reliability and flexibility.

The design presented is an overview of the system required for the tritium recovery system for a lithium module on an ITER type reactor. The design for larger units, sectors, etc. would require an increase in the size of the tritium system but could use the information developed in this report.

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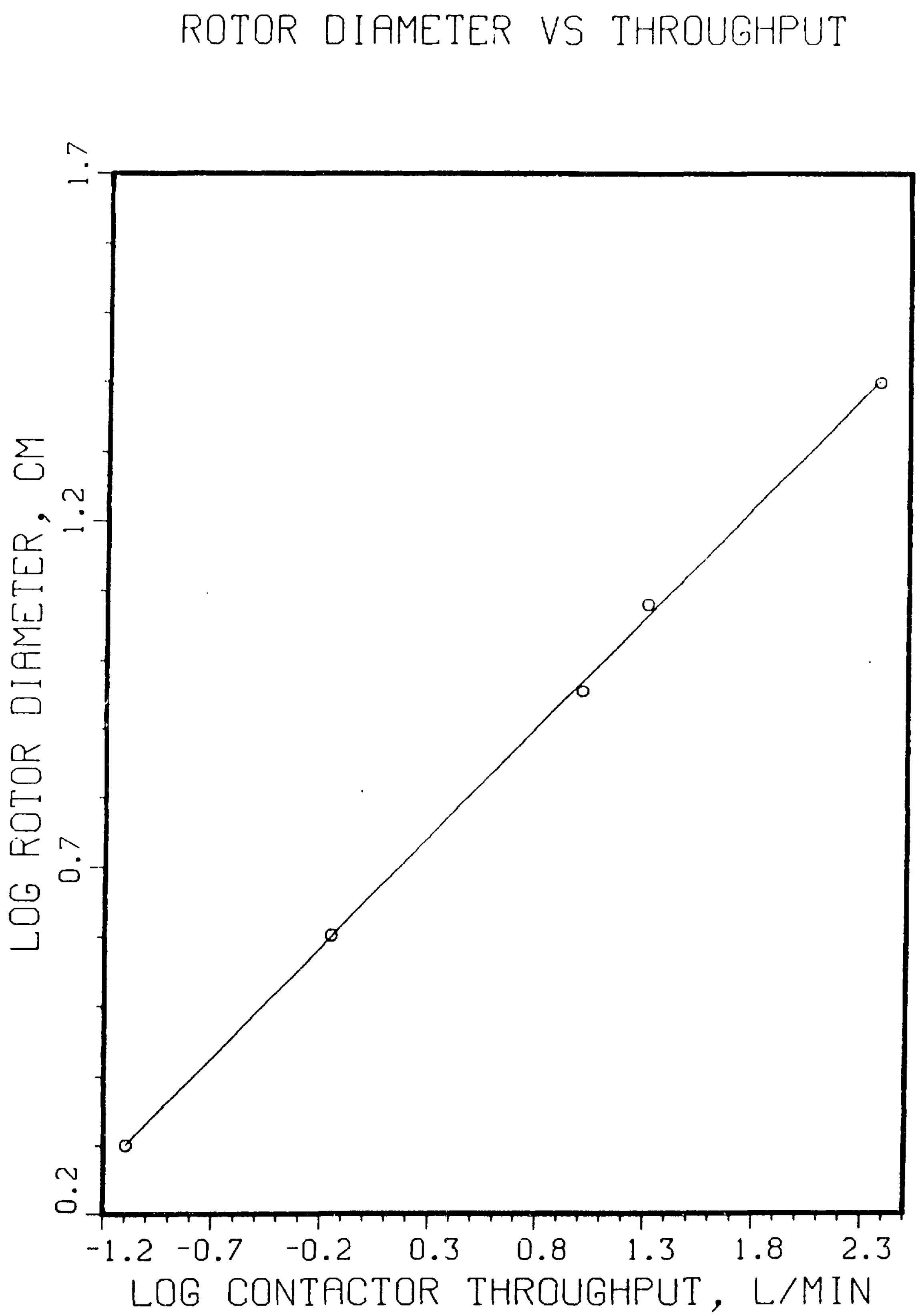

Fig. 1. Plot of Contactor Throughput as a Function of Rotor Diameter.

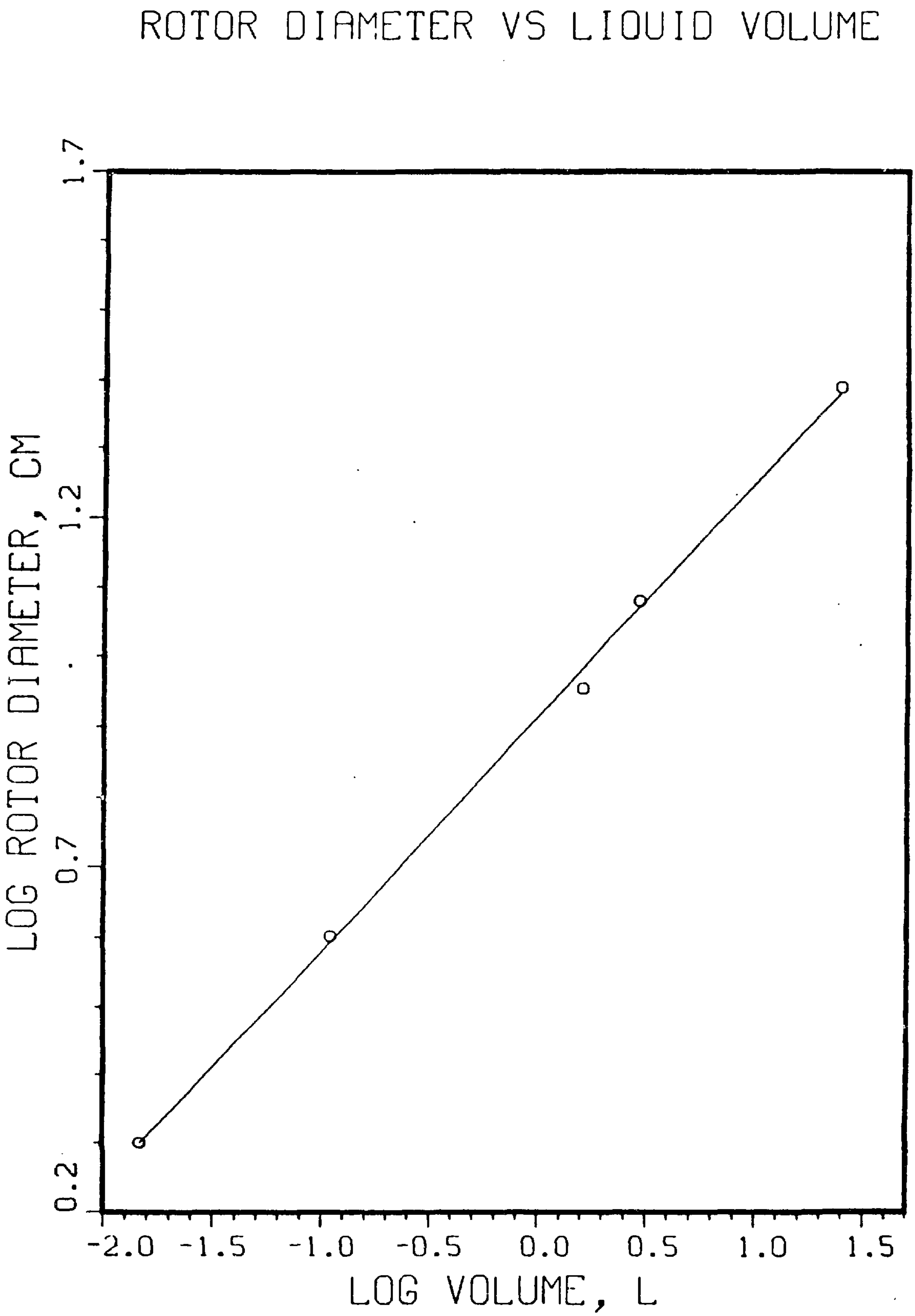

Fig. 2. Plot of Contactor Volume as a Function of Rotor Diameter.

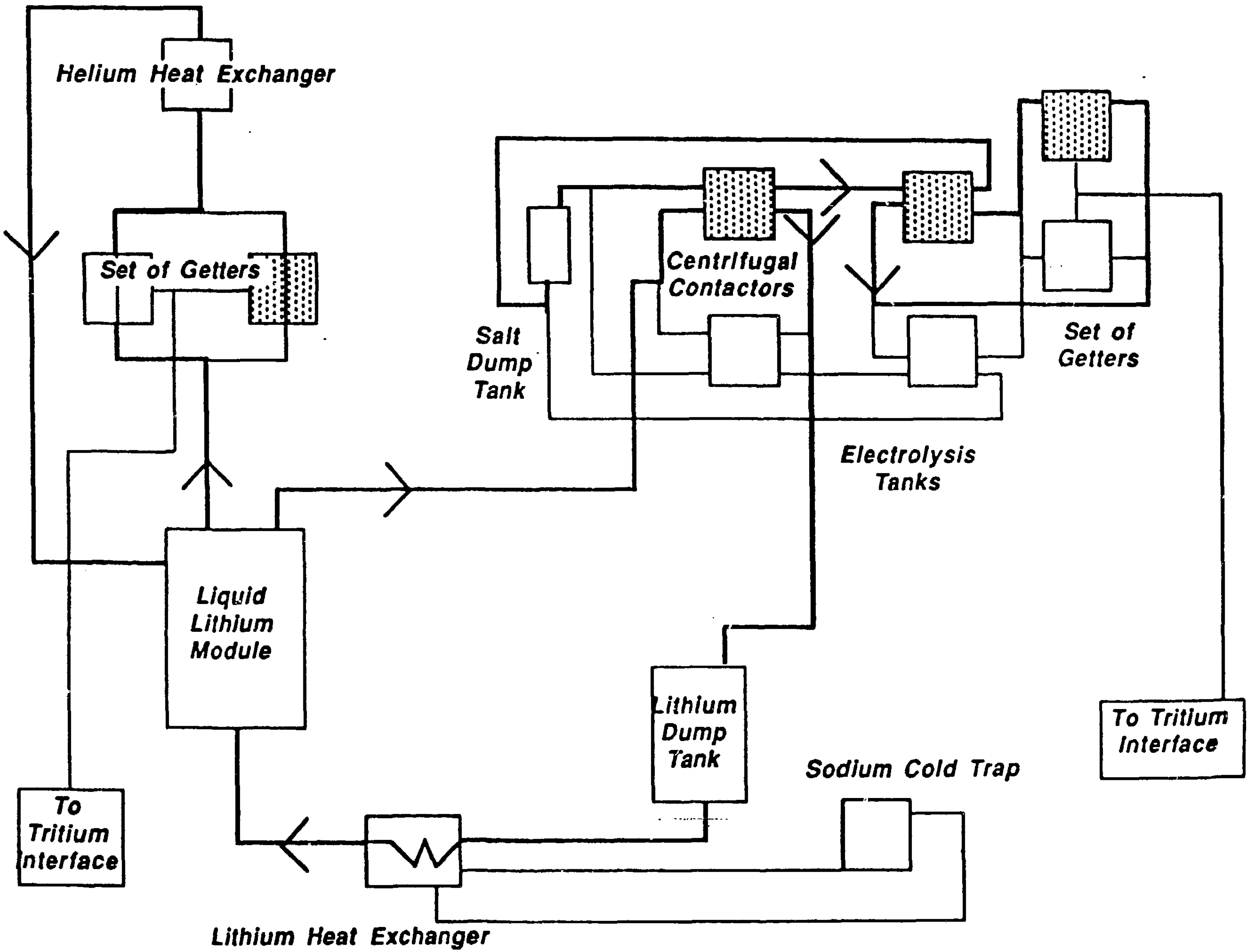

Fig. 3. Tritium Systems for Liquid Lithium Test Module.