CONF-900505--7

DE90 013678

The TEXTOR Helium Self-Pumping Experiment: Design, Plans, and Supporting Ion-Beam Data on Helium Retention in Nickel*

J.N. Brooks,[1] R.E. Nygren,[2] K.H. Dippel,[3] B.L. Doyle,[2] K.H. Finken,[3] Y. Hirooka,[4] A. Krauss,[1] R.F. Mattas,[1] R.T. McGrath,[2] D.L. Smith,[1] and D. Walsh[2]

[1]Argonne National Laboratory, Argonne, IL, USA
[2]Sandia National Laboratories, Albuquerque, NM, USA
[3]IPP Forschungszentrum Jülich Association Euratom-KfA, Jülich, FRG
[4]University of California, Los Angeles, CA, USA

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

MASTER

Manuscript submitted to the 9th International Conference on Plasma Surface Interactions in Controlled Fusion Devices, Bournemouth, UK, May 20-25, 1990.

The TEXTOR Helium Self-Pumping Experiment: Design, Plans, and Supporting Ion-Beam Data on Helium Retention in Nickel*

J.N. Brooks,[1] R.E. Nygren,[2] K.H. Dippel,[3]
B.L. Doyle,[2] K.H. Finken,[3] Y. Hirooka,[4]
A. Krauss,[1] R.F. Mattas,[1] R.T. McGrath,[2] D.L. Smith,[1] and D. Walsh[2]

[1]Argonne National Laboratory, Argonne, IL, USA
[2]Sandia National Laboratories, Albuquerque, NM, USA
[3]IPP Forschungszentrum Jülich Association Euratom-KfA, Jülich, FRG
[4]University of California, Los Angeles, CA, USA

Abstract

A proof-of-principle experiment to demonstrate helium self-pumping in a tokamak is being undertaken in TEXTOR. The experiment will use a helium self-pumping module installed in a modified ALT-I limiter head. The module consists of two, ~25x25 cm^2 heated nickel alloy trapping plates, a nickel deposition filament array, and associated diagnostics. Between plasma shots a coating of ~50 Å nickel will be deposited on the two trapping plates. During a shot helium and hydrogen ions will impinge on the plates through a ~3 cm wide entrance slot. The helium removal capability, due to trapping in the nickel, will be assessed for a variety of plasma conditions.

In support of the tokamak experiment, the trapping of helium over a range of ion fluences and surface temperatures, and detrapping during subsequent exposure to hydrogen, were measured in ion beam experiments using evaporated nickel surfaces similar to that expected in TEXTOR. Also, the retention of H and He after exposure of a nickel surface to mixed He/H plasmas has been measured. The results appear favorable, showing high helium trapping (~10-50% He/Ni) and little or no detrapping by hydrogen. The TEXTOR experiment is planned to begin in 1991.

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

Introduction

Helium removal is a major design and performance issue for next-generation, D-T burning, long pulse tokamaks. Self-pumping impurity control, using certain deposited metals (Ni, V, etc.) to trap helium in-situ, has been identified as a potentially attractive alternative to pumped divertors or limiters [1-3]. Self-pumping can, in principle, eliminate the need for most of the vacuum pumping system, reduce tritium processing and inventory, reduce cost, and possibly achieve better helium removal [1-3].

There is, at present, an encouraging but limited data base regarding helium trapping properties of potential self-pumping materials. In addition to further materials testing, a tokamak test is required prior to consideration of self-pumping for future devices such as ITER. A cost-effective test in TEXTOR is possible because of the use of surplus ALT-I limiter equipment together with excellent plasma helium diagnostics and the extensive plasma heating and operating options available.

The planned experiment is conceptually simple but reproduces key conditions of reactor applications of self-pumping. These are: (1) fairly energetic particle impingement energies (~50-200 eV) into the trapping surface, (2) an oblique incidence magnetic field geometry (and the resulting sheath structure), and (3) high hydrogen/helium ratios and recycling conditions.

The test module is designed to accommodate the expected heat, particle, and disruption loads, and to be compatible with overall tokamak operation. Calculations based on data from the supporting ion-beam experiment, and previous experiments, indicate that the module should be capable of removing a substantial fraction of a 10% plasma helium content during a ~3 sec shot.

Self-pumping concept

The self-pumping concept was proposed [1] as a means of simplifying impurity control in a fusion reactor. The basic concept is to remove helium in-situ by trapping in freshly deposited metal surface layers of a limiter or divertor. The trapping material would be added to the surface at an average rate of 3-5 times the α-production rate to maintain an effective trapping surface. Trapping material can be added, for example, by injecting pellets or exposing rods etc. to a plasma entering a small slot region under a divertor plate [3]. Helium entering the slot region would tend to be trapped on deposited metal surfaces, while hydrogen would recycle back to the main plasma.

A key requirement is for the deposited material to trap helium much better than hydrogen. It has been demonstrated that nickel both traps helium [4] and preferentially traps helium [5]. Several other materials (iron, vanadium, niobium, molybdenum, and tantalum) are also believed to be capable of preferential trapping. The selective trapping in these metals is a result of the negligible solubility of helium in the lattice. Hydrogen, on the other hand, remains in solid solution until it escapes from the surface.

Test concept

The TEXTOR self-pumping experiment will use a small helium self-pumping module, HEMOD, installed in a modified ALT-I limiter head; this is shown schematically in fig. 1. The head is part of a much larger ALT-I assembly [6]. In the ALT-I experiment [6] ions flowing into the slot were neutralized on a small plate and a portion were deflected to an adjacent vacuum pumping duct. The replacement module consists of two heated trapping plates, a nickel evaporation unit, and several probes and a filament temperature diagnostic (not shown). The pumping apparatus of ALT-I is not used, except for a low-conductance path for diagnostic purposes. Plasma ions entering the slot impinge on a trapping/neutralizer plate which is shaped to spread the heat and particle loads. An approximately equal flux of energetic reflected neutrals will impinge on both plates.

Between plasma shots a coating of ~50 Å nickel will be deposited on the trapping plates by the nickel evaporation filaments. During nickel deposition the slot will be blocked by an existing slot closure mechanism to prevent external contamination by evaporated nickel. It is also anticipated that any outside nickel contamination occurring, e.g., by sputtering, can be covered up if necessary by the TEXTOR boronization process [7].

After recoating, the plasma will be run with an initial helium content, in the range of ~1-10%. Wall conditioning shots will likely be required to reduce spurious helium removal effects. Plasma helium content will be monitored by a core plasma charge exchange spectroscopy diagnostic and by helium lines at the limiter [8]. Post-tokamak test analysis of the trapping plates will be performed to assess chemical and structural properties of the trapping surfaces.

A variety of plasma heating and configuration options exist regarding power loading to HEMOD. There is, at present, a total of about 3.4 MW of neutral beam and 4 MW of ICRH power capability. ALT-I can be operated alone for ohmic heating or low power RF/NBI, or in combination with the ALT-II toroidal belt limiter [9] for high power RF/NBI. In the latter case, it is anticipated that reasonably good control of power loading to HEMOD can be achieved by operating ALT-I at different positions in the shadow of ALT-II.

Helium removal will be assessed as a function of plate temperature, helium fraction, and plasma operating conditions.

Collector plate design

The retrofitted system within the ALT-I head, shown in detail in fig. 2 includes the two heated collector plates and a filament array for nickel deposition consisting of two sets of five tungsten conductors, with nickel wire wrap or coating, supported between bus bars. The base plates will be constructed of a non-magnetic alloy such as Inconel, with (jacketed) Thermocoax heaters embedded within a plasma sprayed nickel alloy backing.

The filament design must ensure reliable contact between tungsten and nickel while preventing embrittlement of the tungsten. Initially, braided filaments of tungsten wires and nickel wires were produced but unreliable thermal contact led to overheating, some melting of the nickel which wetted the tungsten and caused embrittlement. Development of other (coated) filament designs is in progress.

A key objective of the design is to maximize the helium trapping by spreading the impinging helium flux evenly over the upper plate. Plasma heating of the plate is a severe constraint for non-ohmic heating operation and it is necessary to limit the total power and spread the flux in order to keep the temperature of the plates within the specified range of 400-550°C. Implicit in this task are (1) shaping to distribute heat and particle loads, (2) a robust yet compact heater design -- some machining of the thin ALT-I head is still necessary to mask the leading edge of the plate, (3) monitoring of the plate temperature, and (4) accurately fabricating and installing a plate that mimics the poloidal field curvature. Although the plate has only poloidal curvature, the angle of field impingement varies significantly along the length due to the change in major radius (see fig. 3). The heat flux is proportional to the product of the exponential decay factor from the scrape-off and the sine of the angle of incidence, which increases along the plate, a feature exploited in design. With a scrape-off length of 2.2 cm [10] and a nominal pitch (at the leading edge) of 5°, the maximum heat load occurs near the center of the plate rather than at the leading edge.

Extensive thermal analysis of the upper plate, necessary for design, was done using the PATRAN and ABAQUS codes which include a 3-D magnetic map of TEXTOR overlayed with the configuration of the collector plate [11]. Variables included scrape-off length, angle of the plate (from the toroidal tangent) and the overall heat load into the ALT-I throat. For respective values of 2.2 cm, 5° and 20 kW, computed surface temperatures on the plate reach fairly uniform values of ~500-550°C.

The heat loads are sufficiently high in a beam heated discharge that operating TEXTOR with ALT-I alone would overheat the upper plate. The allowable total power into the ALT-I throat is in the range of 20-25 kW. As mentioned, this limitation will constrain experiments with ALT-I alone to ohmic or low NBI/RF discharges. Power sharing with ALT-I shadowed by ALT-II will be used for most beam heated shots.

A summary of the module design and estimated performance is shown in table 1.

Data on helium trapping in a deposited nickel layer

To obtain data on saturation values for helium retention in nickel layers for the helium self-pumping experiment, samples prepared at Argonne National Laboratory were exposed to helium plasmas in PISCES [12], a plasma source at UCLA. The samples consisted of nickel films that were deposited on a pure Ni substrate at room temperature using a laboratory mock-up of the TEXTOR experiment. Disks one inch in diameter and 1/16" thick were placed on the PISCES target holder and implanted with He^+ at energies of 50 eV and 140 eV. Ion fluxes ranged from 2.5×10^{17} cm^{-2} sec^{-1} for the 140 eV implantation, and from 8.2×10^{17} to 1×10^{18} cm^{-2} sec^{-1} at 50 eV. Assuming implantation depths of 25 Å and 13 Å respectively, the implantation zone is saturated in less than 100 msec at these fluxes. Since heating was provided by the incident plasma, it was necessary to continue bombardment for several hundred seconds to obtain sample temperatures in the range 100-575°C. The time required to remove a 13-25 Å layer is only about ten seconds, so the surface was implanted and resputtered many times during the heating cycle. The samples therefore represent a "snapshot" of the He saturation during approximately the last ten seconds, and are therefore characteristic of the temperature just before the plasma was terminated. Since plasma heating was used to control the temperature, those samples which went to the highest temperatures also received the highest fluence. Measurements of retained helium were performed at Sandia National Laboratory using ERD (Elastic Recoil Detection) analysis with 16 MeV Si. The data are shown in table 2.

Based on a flat surface with He implanted uniformly to a depth equal to the maximum range (including half the straggling) of 13 Å and 25 Å for 50 eV and 140 eV, respectively, the calculated values for trapping (He as atomic % of Ni) are 10-12% for the 140 eV samples. The 50 eV samples cover the range of 5-48% and are proportional to the fluence received. These data confirm an important point -- at temperatures around 500°C with relatively low energy implantation, the saturation for He trapping reaches fairly high levels. The 50 eV data are provocative in their apparent fluence (or temperature) dependence; further laboratory experiments are planned to provide additional data on fluence and temperature effects.

Samples exposed to PISCES were also subjected to subsequent "washing" with implanted hydrogen to see if the implanted He was retained. Tests were conducted using a Colutron ion beam implanter at Sandia. At 500°C with ion beam doses at 200 eV of up to 2×10^{17} H/cm^2 samples with an initial areal density of 6.2×10^{15} He/cm^2 showed, to within experimental error, no loss of helium. This suggests that the simultaneous hydrogen flux in TEXTOR will have little effect on He retention. Data by Pontau et al. [5] with 4 keV He and 2 keV H showed a similar lack of effect by hydrogen.

Conclusions

A helium self-pumping experiment has been designed for TEXTOR and is currently being fabricated. The self-pumping module can operate alone for OH shots and in the shadow of ALT-II for full power NBI and RF shots. Experimental laboratory results on He implanted prototype nickel deposited surfaces appear indicative of a successful demonstration of helium removal in TEXTOR. Whatever the outcome, however, the TEXTOR experiment should provide a useful indication of whether to proceed with further development of this potentially important concept, for tokamak reactors.

Acknowledgments

The authors wish to acknowledge extensive design work done by Hank Caldwell (EG&G) and thermal analyses performed by Hank Caldwell and Frank Dempsey (EG&G).

References

[1] J.N. Brooks and R.F. Mattas, J. Nucl. Mater. 121 (1984) 392.

[2] J.N. Brooks et al. in: Proc. 12th Symposium on Fusion Engineering [IEEE Catalog Number 87CH2507-2 (1987) 235].

[3] J.N. Brooks, R.F. Mattas, D.L. Smith, and A.M. Hassanein, Self-Pumping Impurity Control Systems for INTOR, Argonne National Laboratory Report, ANL/FPP/TM-212.

[4] D.J. Reed, F.T. Harris, D.G. Armour, and G. Carter, Vacuum 24 (1974) 179.

[5] A.E. Pontau, W. Bauer, and R.W. Conn, J. Nucl. Mater. 93&94 (1980) 564.

[6] K.H. Dippel et al., J. Nucl. Mater. 145-147 (1987) 3.

[7] Y. Winter et al., J. Nucl. Mater. 162-164 (1989) 713.

[8] K.H. Finken et al., Helium Exhaust Studies with the ALT-II Pumped Limiter in TEXTOR, 9th PSI Conference (to be published in J. Nucl. Mater.).

[9] D.M. Goebel et al., J. Nucl. Mater. 162-164 (1989) 115.

[10] R. McGrath, Thermal Loads on Tokamak Plasma Facing Components During Normal Operation and Disruptions, Vac. Tech. & Applied Ion Physics, to be published (also SAND89-2064, January 1990).

[11] J. Koski, F. Dempsey, and R. McGrath, Contouring of an Inertially Cooled Pump Limiter for Tore Supra, submitted to Fusion Engineering and Design (also SAND89-2994J, to be published).

[12] Y. Hirooka et al., A New Plasma-Surface Interactions Research Facility: PISCES-B and First Material Erosion Experiments on Boronized Graphite. JVST-A to be published.

Table 1. Self-pumping module design summary and estimated performance parameters.

Parameter	Value
1. Trapping plates structural material	INCONEL
2. Helium trapping material	NICKEL
3. Trapping surface temperature	400-550°C
4. Trapping area (~25 cm x 25 cm x 2)	~0.125 m^2
5. Recoating depth, per shot	50 Å
6. Helium trapping capability, per shot[1]	~5×10^{18} He atoms
7. Plasma temperature, near plates ($T_{e_o} = T_{i_o}$)	15-25 eV
8. He^{+2} impingement energy[2]	135-225 eV
9. D^+ impingement energy[2]	75-125 eV
10. Transport power to trapping/neutralizer plate (maximum)	20 kW
11. He^{+2} current to trapping/neutralizer plate[2-3]	~8.0×10^{19} s^{-1}
12. Energetic ($\geq$ 50 eV) neutral helium (He) current to both plates	~8.0×10^{19} s^{-1}
13. Initial plasma helium content[4]	~1.0×10^{19} He atoms
14. Time constant for helium removal[4,5]	0.4 s

(1) For 20% He/Ni trapping ratio.

(2) For sheath potential $\left|\frac{e\phi}{kT_{e_o}}\right| = 3$

(3) For 10% helium plasma, 20 kW transport power, T_{e_o} = 20 eV

(4) For 10% helium plasma, core deuterium density, $N_D = 1.5 \times 10^{19}\ m^{-3}$

(5) For 20% helium trapping probability (per impingement) and equal helium and hydrogen particle confinement times.

Table 2. He retention in Ni layers exposed to He plasma in PISCES

Energy/Sample		Temp. (°C)	Retained He (He/cm²)	Time (s)	Flux (A/cm²)	Fluence (He/cm²)
140 eV	2	100	2.845E+15	214	0.045	6.0E+19
	3	175	3.089E+15	51	0.084	2.7E+19
	4	275	3.145E+15	87	0.085	4.6E+19
	5	350	2.741E+15	116	0.084	6.1E+19
	6	425	3.054E+15	174	0.085	9.2E+19
	7	500	3.089E+15	187	0.088	1.0E+20
	8	575	2.939E+15	230	0.088	1.3E+20
50 eV	9	100	7.168E+14	39	0.110	2.7E+19
	10	175	2.189E+15	93	0.110	6.4E+19
	11	275	4.613E+15	167	0.140	1.5E+20
	12	350	4.194E+15	162	0.150	1.5E+20
	13	425	4.089E+15	188	0.160	1.9E+20
	14	500	4.719E+15	218	0.140	1.9E+20
	15	575	6.838E+15	293	0.150	2.7E+20

FIGURE CAPTIONS

Fig. 1. Test concept schematic.

Fig. 2. Self-pumping module design: ALT-I midplane section.

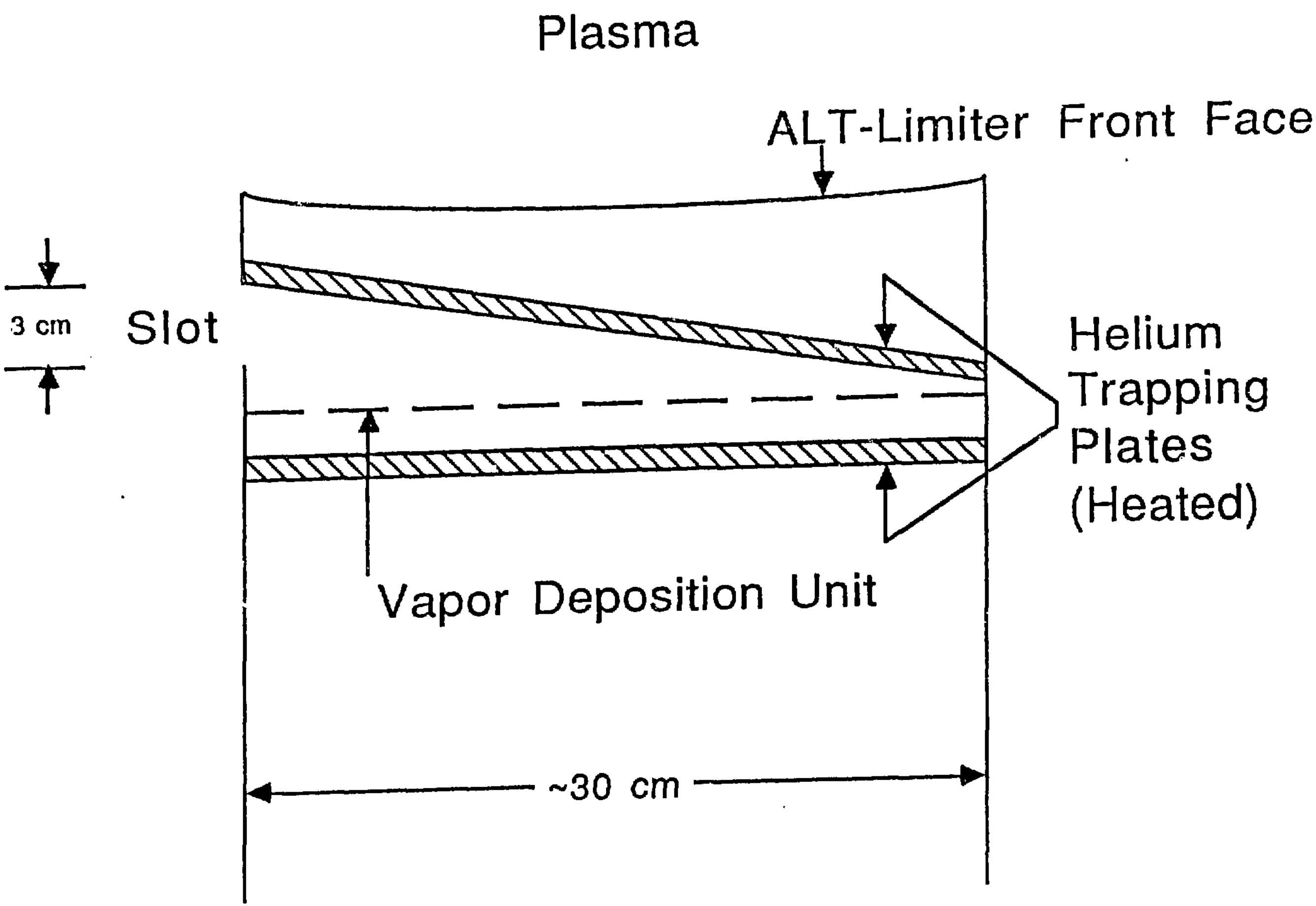

Fig. 1. Test concept schematic.

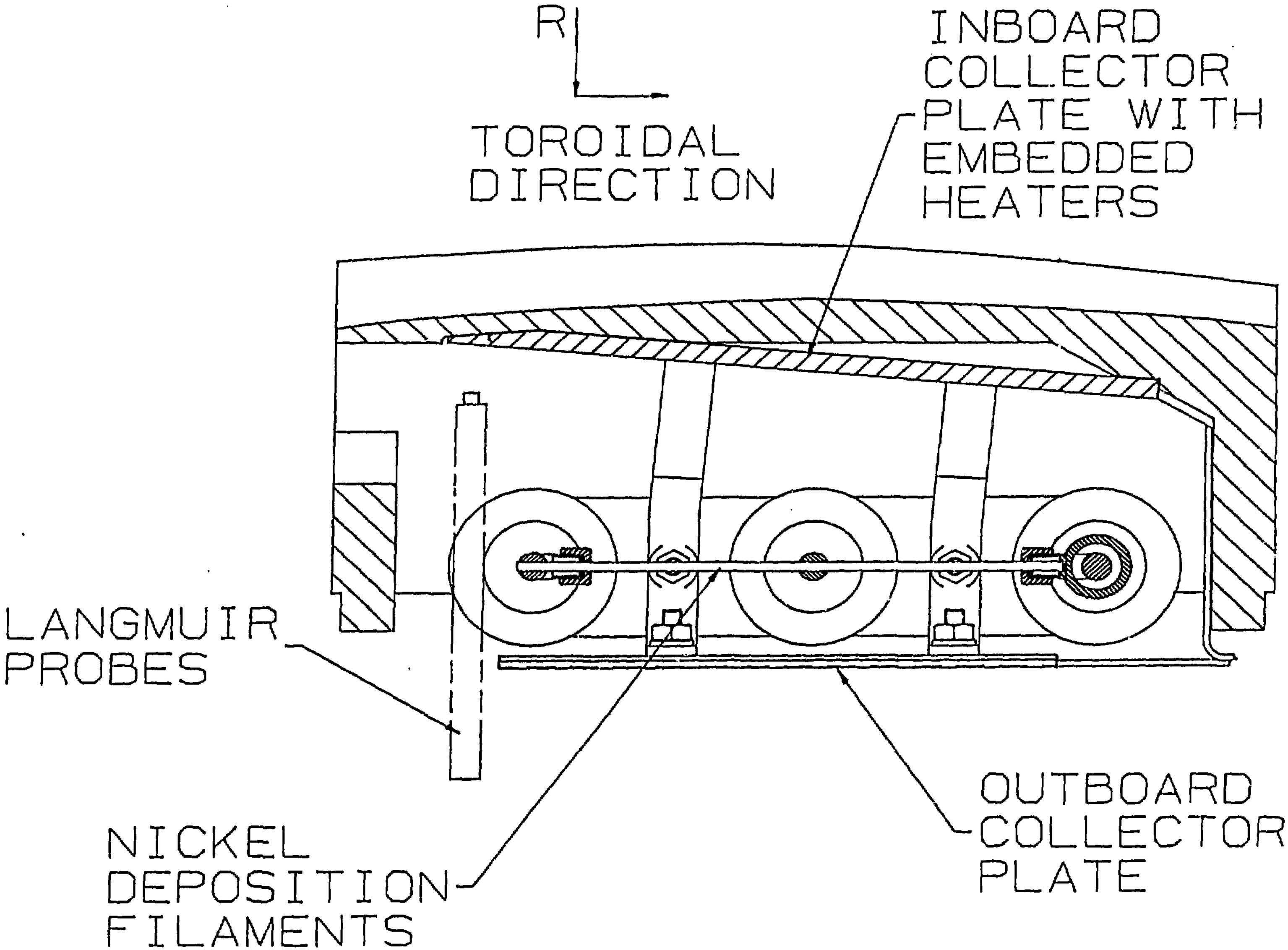

Fig. 2. Self-pumping module design: ALT-I midplane section.