

DESIGN OF A HELIUM-COOLED MOLTEN-SALT FUSION BREEDER*

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ABSTRACT

A new conceptual blanket design for a fusion reactor produces fissile material for fission power plants. Fission is suppressed by using beryllium, rather than uranium, to multiply neutrons and also by minimizing the fissile inventory. The molten-salt breeding media ($\text{LiF}+\text{BeF}_2+\text{ThF}_4$) is circulated through the blanket and on to the online processing system where ^{233}U and tritium are continuously removed. Helium cools the blanket including the steel pipes containing the molten salt. Austenitic steel was chosen because of its ease of fabrication, adequate radiation-damage lifetime, and low corrosion rate by molten salt. We estimate the breeder, having 3000 MW of fusion power, produces 6400 kg of ^{233}U per year, which is enough to provide make up for 20 GWe of LWR per year (or 14 LWR plants of 4440 MW_t) or twice that many HTGRs or CANDUs. Safety is enhanced because the afterheat is low and the blanket materials do not react with air or water. The fusion breeder based on a pre-MARS tandem mirror is estimated to cost \$4.9B or 2.35 times an LWR of the same power. The estimated present value cost of the ^{233}U produced is \$40/g if utility financed or \$16/g if government financed.

The present design evolved from earlier concepts^{1,2,3} with changes avoiding the problems of prior designs. For example, steel was substituted for molybdenum and graphite coatings were eliminated. The beryllium, rather than being in large logs, is now made of small radiation damage resistant spheres. This paper is a summary of a large report on the subject.⁴ The fusion neutron source for this design is the tandem mirror, but equally well could have been other fusion concepts such as

*Work performed under the auspices of the U.S. Department of Energy by the Lawrence Livermore National Laboratory under contract number W-7405-ENG-48.

the tokamak. The particular tandem mirror design is based on a pre-MARS design⁵. The plant parameters are given in Table I. The technologies used are listed in Table II.

Table I. Plant parameters.

P_{nuclear}	4440 MW
P_{fusion}	3000 MW
$P_{\text{alpha particle}}$	600 MW
P_{blanket}	3840 MW
P_{electric}	1380 MW _e
$P_{\text{wall load}}$	2 MW/m ²
Length of blanket	127 m
First wall radius	1.5 m
F_{net}	0.6
M^{a}	1.6
Fissile production	6380 kg ^{233}U /yr at 80% capacity factor
Total cost	\$4867M

^a F_{net} is fissile atoms bred/tritium consumed; M is the energy released in the blanket per tritium consumed divided by 14 MeV.

BLANKET DESIGN.

The blanket concept uses high-pressure helium as the coolant and beryllium spheres as the neutron multiplier. All fertile material for breeding both tritium and fissile fuel is in the form of molten fluoride salts of lithium and thorium that flow slowly through an array of tubes (see Figs. 1 and 2).

The molten-fluoride salt, whose composition is given in Table II, is stable to both thermal and radiation decomposition because of the rapidity of recombination. Corrosion rates of iron-based alloys are low when the salt is maintained in a reducing state. A similar salt, with different mole fractions, was used at Oak Ridge National Laboratory in a molten-salt reactor that operated successfully for several years. The uranium produced by neutron

Table II. Technologies.

Blanket coolant	Helium
Structural material	316 type st. steel
Neutron multiplier	Beryllium pebbles
Tritium and fissile breeder	Molten salt: LiF 70 mol% BeF ₂ 12 mol% ThF ₄ 18 mol%
Neutron reflector	SiC

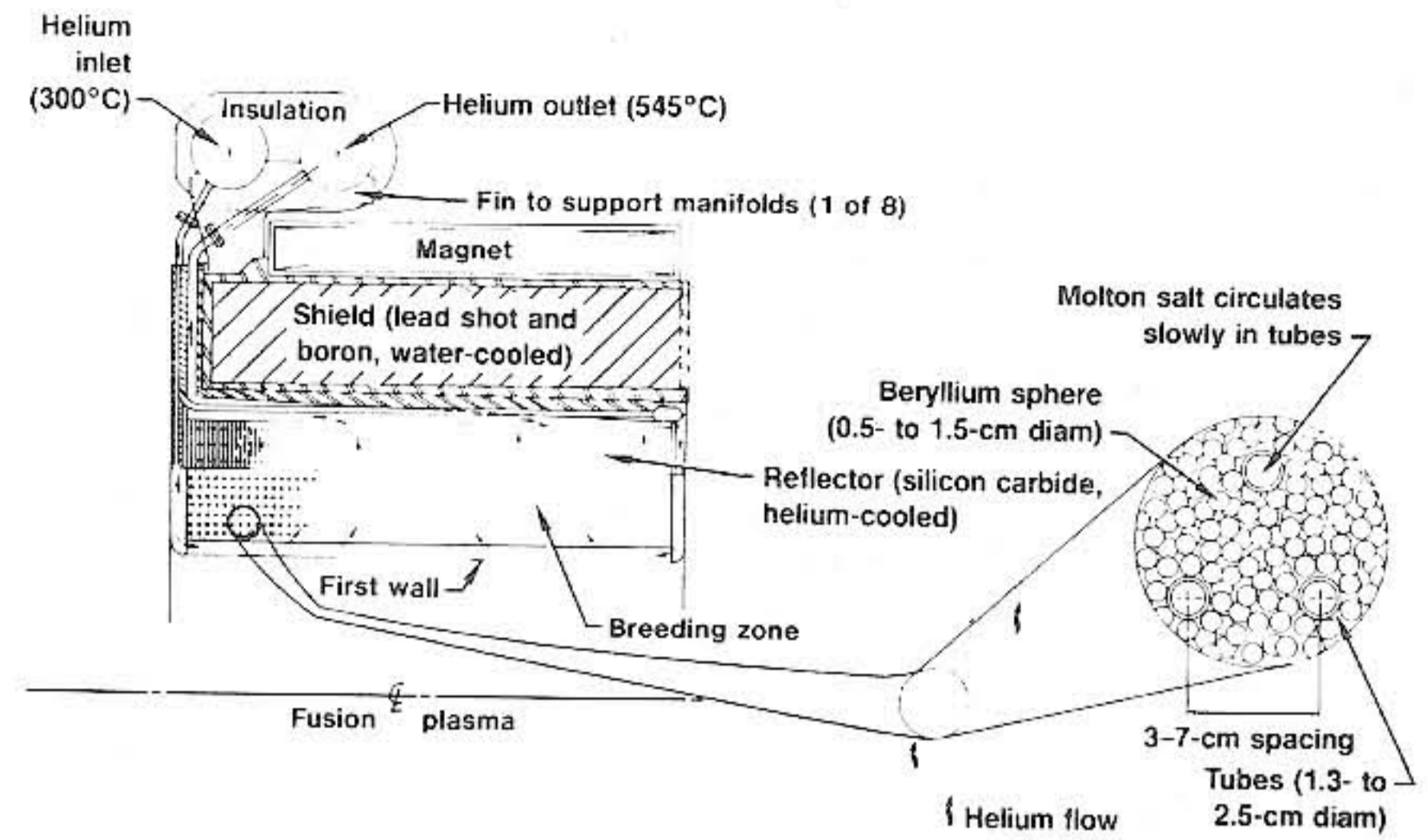


Figure 2. Cross section along the axis of one segment of a helium-cooled molten-salt blanket, showing arrangement of helium flow and of beryllium spheres and tubing for the molten salts.

Table III. Each neutron multiplication reaction in beryllium yields two neutrons and two alpha particles (helium nuclei). Most of these helium atoms remain trapped in the beryllium lattice, taking up space but not contributing to the bonding that holds the lattice together. The resulting stress that causes the beryllium spheres to expand would eventually break them apart if it were allowed to continue.

Because any break-up of the beryllium spheres would yield smaller particles that could plug the channels through which the helium flows and thus produce hot spots in the blanket, it is important to remove the beryllium spheres periodically before this can happen. We have verified experimentally that the spheres will

Table III. Beryllium requirements for the molten-salt blanket.

Total blanket volume ^a	860 m ³
Pebble volume ^b	470 m ³
Pebble quantity ^c	890 x 10 ⁶
Pebble mass	0.96 g
Beryllium mass ^d	860 MT
Average beryllium lifetime	5 yr
Annual pebble throughput	180 x 10 ⁶ /yr
Annual beryllium mass throughput	170 MT/yr

^a 127-m central cell, 3000 MW fusion power, 0.6-m-thick blanket starting at 1.5-m radius, and a 2 MW/m² wall load.
^b 10% of blanket volume is tubes; 60% packing in remainder of blanket.
^c 1-cm-diameter pebbles.
^d 1.84 g/cm³ beryllium density.

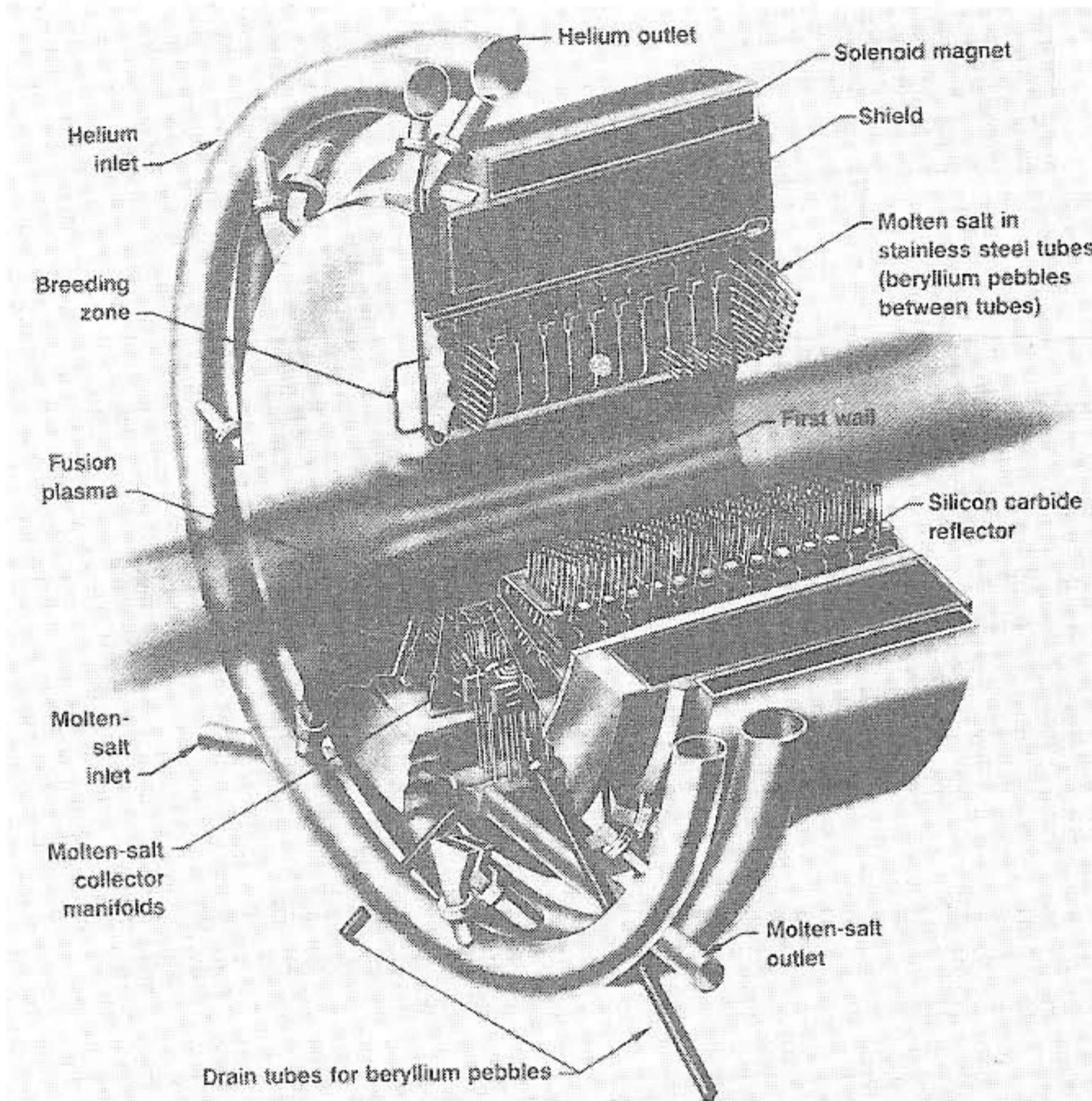


Figure 1. One module of a helium-cooled molten-salt blanket. Helium under 5-MPa pressure flows from the rearmost ring header to the apex of each pod, then radially outward through the blanket to the forward ring header, and thence to heat exchangers for generating electricity.

transmutation of ThF₄ is in the form of UF₄. Upon reacting with excess fluorine in an external tank, the UF₄ is converted to UF₆, which is volatile and easily separated. This process is called fluorination. UF₆ is a standard form of uranium used in the industry.

To acquire a high enough heat capacity to cool the blanket efficiently, helium must be at a pressure of 5 MPa or higher. Helium neither absorbs neutrons nor promotes self-welding reactions that might interfere with the free flow of the beryllium spheres as they are removed for remanufacturing. Some parameters describing the blanket and the beryllium pebbles are given in

flow freely, without bridging or blocking, if they are no larger than half the space between the tubes. To avoid any possible obstruction because of swelling of the beryllium, we made the spheres 1 cm in diameter (one-third the distance between the tubes).

Because the fertile material is constantly circulating, it is possible in this design to use a continuous processing system to recover bred tritium and uranium-233 from the molten salts. We selected fluorination, which removes uranium-233 but leaves protoactinium and many of the fission products.⁶ More complex processes that do remove fission products, such as reductive extraction and metal-transfer treatments, turn out to be unnecessary.⁷ Continuous processing allows the amount of uranium-233 to be kept low to minimize unwanted fission reactions.

In addition to its many advantages, this blanket design has a few disadvantages. One is that the flat side walls of each module are not stiff enough to support the internal pressure of 5 MPa. Instead, they must be made thin, each one being supported by the adjacent wall pressing in from the other side. This means that all the modules are interdependent; none can be removed for service without depressurizing the entire system. Another consequence of this enormous pressure that forces adjacent module walls together is the possibility that they may self-weld after a time. If this happened, it would be impossible to remove one module without causing damage to the other. To prevent this, it may be necessary to insert thin ceramic spacers of alumina to keep the walls apart.

NUCLEONICS

Breeding performance is predicted with the aid of Monte Carlo transport codes (see Table IV). The modeling of the blanket takes into account the heterogeneous nature of the blanket, the effect of structural material and resonant self-shielding. The uncertainty, because of the combined effects of modeling, nuclear data and methods, is guessed to be as high as $\pm 25\%$ for F_{net} . Breeding is sensitive to the amount of structural material, mostly Fe. By increasing the steel structural fraction in the blanket from 3% to 5%, the net

Table IV. Calculated breeding performance.

T_a	1.0
F_{net}^a	0.6
E (MeV)	22.4
M (E/14)	1.6

^aAtoms bred per triton consumed.

breeding goes from 0.65 to 0.61. Because of the low atom fraction of thorium and ^6Li in the blanket, parasitic neutron capture in structural material mandates low structural fraction blanket designs. Recent and ongoing work on the nuclear data on beryllium suggest that the old data resulted in an overestimate of net breeding. An experiment underway at the Idaho National Engineering Laboratory using the manganese bath method to measure neutron multiplication and an experiment recently completed at LLNL, should reduce this uncertainty substantially.

THERMAL HYDRAULICS

The amount of heat generated in the salt, in the tube walls, and in the beryllium is calculated by neutron and gamma-ray transport codes at several radial locations. Heat-transfer coefficients are estimated for tubes with roughened surfaces in a pebble bed, based on separate data from close-packed tubes (cross-flow) and from pebble beds ($h = 0.2 \text{ W/cm}^2\text{C}$). Then the temperature of the salt-steel interface of the tubes is calculated (see Fig. 3). As indicated in Fig. 3, tube sizes can vary from the front to the back of the blanket. The inlet helium temperature is 300°C and the outlet is 545°C . The calculated maximum steel temperature is only slightly over 550°C . The concern over temperature variations around each tube and thus hot spots is largely alleviated by the averaging effect of the flowing salt. The salt is force-circulated at about 0.1 ms^{-1} and internally convects at a similar speed because of gravity and a density gradient caused by the temperature gradient in the salt. We have considered the pros and cons of having a frozen salt layer on the inside of the tube (i.e., tube temperature kept below the salt freezing point) and generally favor the no-frozen layer case. We recommend heat-transfer measurements be carried out in a pebble bed with pipes and helium crossflow to reduce the uncertainty in predicting temperatures at various places.

A start-up scenario was devised for situations where the inlet helium temperature (300°C) is substantially below the melting temperature of the salt (530°C). Before the fusion reactions are turned on, the helium is circulated through the blanket after going through a preheater, thus raising the entire blanket to the 545°C inlet temperature, which is identical to the operating outlet temperature. Preheated molten salt is then introduced into the blanket. As the fusion reaction rate is brought up, the helium circulation rate is adjusted and the preheater turned down. Figure 4 shows the start-up conditions. Shutdown is accomplished by draining the salt and reducing the helium circulation rate. Freeze-up is easily prevented by the salt's heat capacity and residual nuclear decay heat. The salt drain tanks are cooled by heat pipes, thus, the emergency cooling is entirely by passive means.

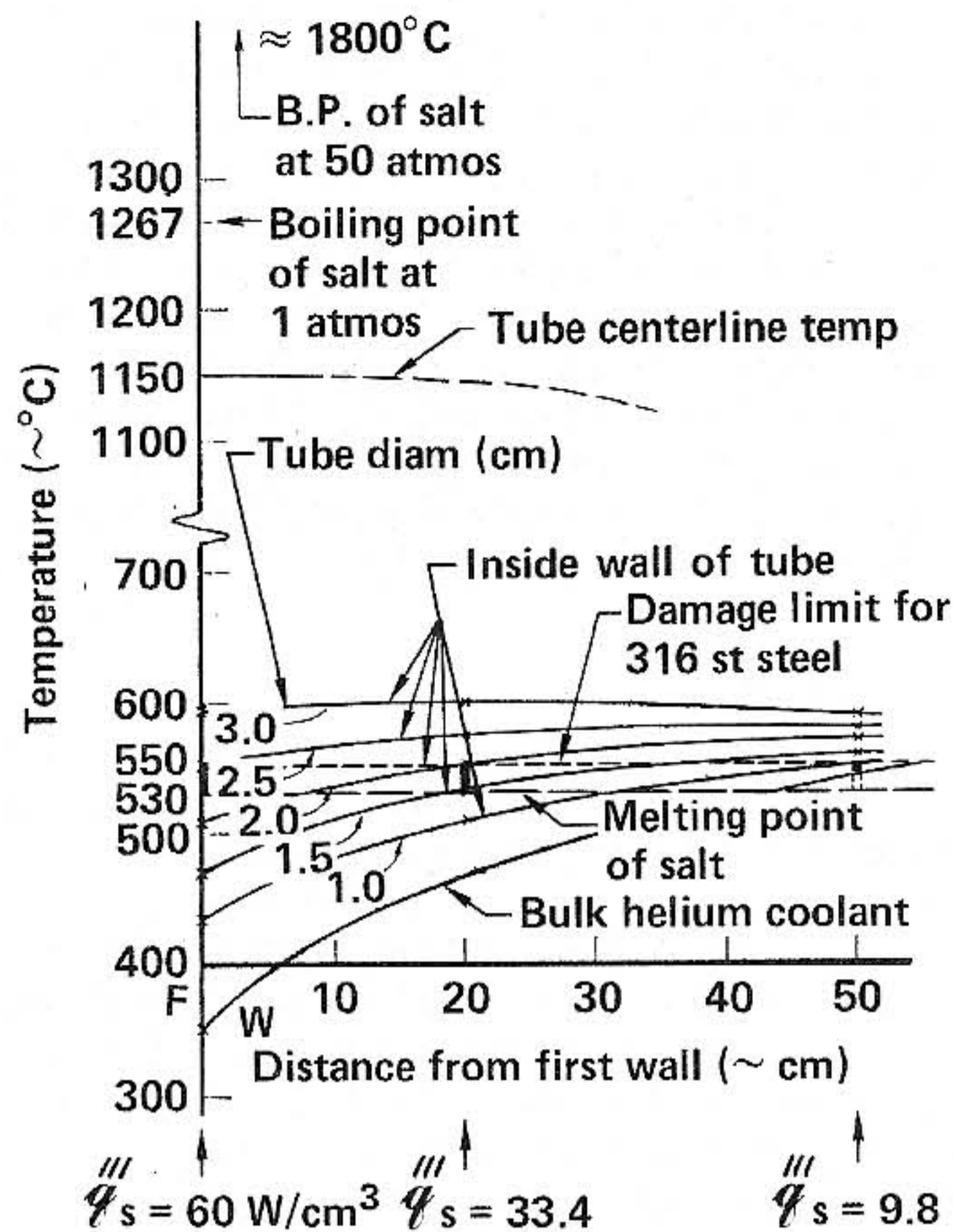


Figure 3. Helium coolant and salt-tube temperature versus position in blanket.

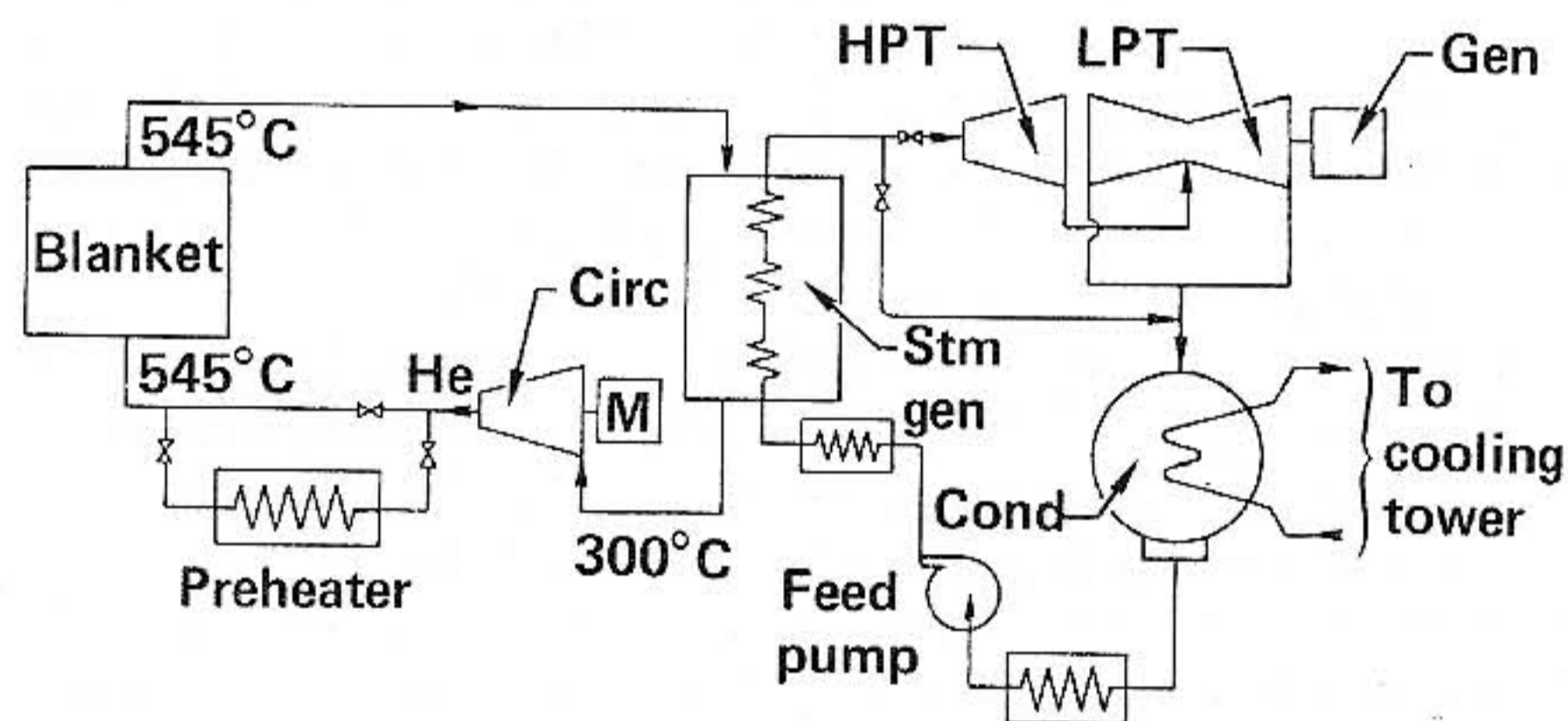


Figure 4. Startup scenario.

MATERIALS SELECTION AND PROPERTIES

From a neutronics point of view, we would like to maximize the ThF_4 concentration; however, higher ThF_4 concentration raises the melting temperature. For the molten-salt reactor experimental work, we used 12% molar ThF_4 . We considered using 27% ThF_4 ; however, the present design uses an intermediate composition of 18% which gives a melting point of 530°C . Some properties of the salt are given in Table V.

We chose austenitic steel (PCA) for this design. The maximum operating temperature is 550°C , chosen to avoid excess helium embrittlement. The lifetime is 100 dpa (about $8 \text{ MW}\cdot\text{yr}/\text{m}^2$ or 5.7 yr at $2 \text{ MW}/\text{m}^2$ and 70% availability), after which time runaway swelling is predicted. There is no minimum temperature, as there is no ductile-brittle transition above room temperature. Also, no special heat treatment of welds is necessary. Corrosion of steel is governed by reaction of the chromium of the steel with the salt and is limited

Table V. Composition and properties of blanket salt.

Composition (mole %):			
LiF	72	70	71
BeF_2	16	12	2
ThF_4	12	18	27
Liquidus ($^\circ\text{C}$)	500	530	565
Properties at 600°C :			
Density (g/cm^3)	3.35	3.87	4.52
Liq. heat capacity ($\text{cal}/\text{g}^\circ\text{C}$)	0.33		0.23
Viscosity (centipoise)	12		15-25
Vapor pressure (torr)	<0.1		--
Thermal conductivity ($\text{W}/^\circ\text{C cm}$)	0.011		0.007
Heat of fusion (cal/g)	63		54
Elec. conductivity ($\Omega\text{-cm}$)	2.12		--
Expansion on melting (vol%)			~ 5

by the oxidation potential of the salt and the solid state diffusion of chromium in the steel. Experiments show that for 316 type steel and a $\text{UF}_4:\text{UF}_3$ ratio of 10 in the salt which is a reducing state, the corrosion will only be $2 \mu\text{m}/\text{yr}$. Steel with less chromium will corrode at a slower rate. Based on both acceptable mechanical properties following high-fluence neutron exposures and acceptable corrosion resistance, we expect 316 steel in contact with fluoride salts containing uranium under controlled oxidation conditions is expected to be feasible.

TRITIUM PERMEATION AND RECOVERY.

A study of tritium permeation and recovery for molten salt for the fusion breeder is reported in Ref. 8. This study assumes tritium to be a gas dissolved in molten salt, with TF formation suppressed. Tritium permeates readily through the hot steel tubes of the reactor and steam generator and will leak into the steam system at the rate of about 1 g/day in the absence of special permeation barriers, assuming that 1% of the helium coolant flow rate is processed for tritium recovery at 90% efficiency per pass. Tritiated water in the steam system is a personnel hazard at concentration levels well below one part per million and this level would soon be reached without costly isotopic processing. Alternatively, including a combination of permeation barriers on reactor and steam generator tubes and molten salt, we estimate that processing will reduce the leak rate into the steam system by over two orders of magnitude. For the option with the lowest estimated leak rate, 55 Ci/d, it may be possible to purge the steam system continuously to prevent tritiated water buildup. Oxygen overpressure in the helium stream to oxidize tritium can further reduce releases. At best,

isotopic separation of dilute tritiated water may not be necessary and for higher leak-rate options the isotopic processing rate can be reduced.

The proposed permeation barrier for the reactor tubes is a 10- μ m layer of tungsten which, in principle, will reduce tritium permeation by a factor of about 300 below the bare-steel rate. The 10- μ m tungsten barrier on the inside of the tubes is made by chemical vapor deposition (CVD). The reaction $WF_6 + 3H_2 \rightarrow W + 6HF$ proceeds when the temperature is raised to the range of 400 to 600° C. The amount of material is so small (65,000 lbs) that the material cost (\$4M) is not important. Stainless steel will require a coating such as nickel to obtain a good bond. The techniques will require further development; however, there is quite a lot of industrial practice in CVD upon which to base this development. A relatively thick 1-mm aluminum sleeve was selected to suppress permeation through the steam generator tubes. This gave a calculated reduction of more than a factor of 500 relative to bare steel, including a factor of 30 because of an assumed oxide layer. This is essentially a brute-force approach that may well be improved upon by the development of more sophisticated permeation barriers.

To gain a better understanding of permeation effects, we derived equations describing steady-state tritium permeation for a multi-layer tube wall. Molten-salt boundary-layer resistance can be important in the absence of a good permeation barrier, or for a low-power tube coated with a nominal 1- μ m tungsten barrier. Permeabilities of various metals are shown in Fig. 5. Examination of the radial flux equation shows a complicated dependence on upstream partial pressure, which reduces to a linear dependence at low pressures where Henry's Law materials become flux limiters and a square-root dependence at high tritium partial pressures where Sievert's Law materials are flux limiting. Permeation barriers are shown in Fig. 6. The barriers are shown on the outside of the tubes but could equally well be on the inside. For the molten-salt tubes, the inside barrier would greatly reduce the tritium inventory in the tube walls and further reduce the already low corrosion rate. The tritium fraction escaping through the tube walls was quantified for limiting cases of Henry's Law and Sievert's Law barriers as flux limiters. The tritium-recovery system flow sheet is shown in Fig. 7. Because of the low solubility of tritium in the reducing salt, a simple flash separator will allow removal of the tritium and other noncondensable gases, mainly helium.

Tritium removal from helium is virtually a standard system. The bulk of the tritium is

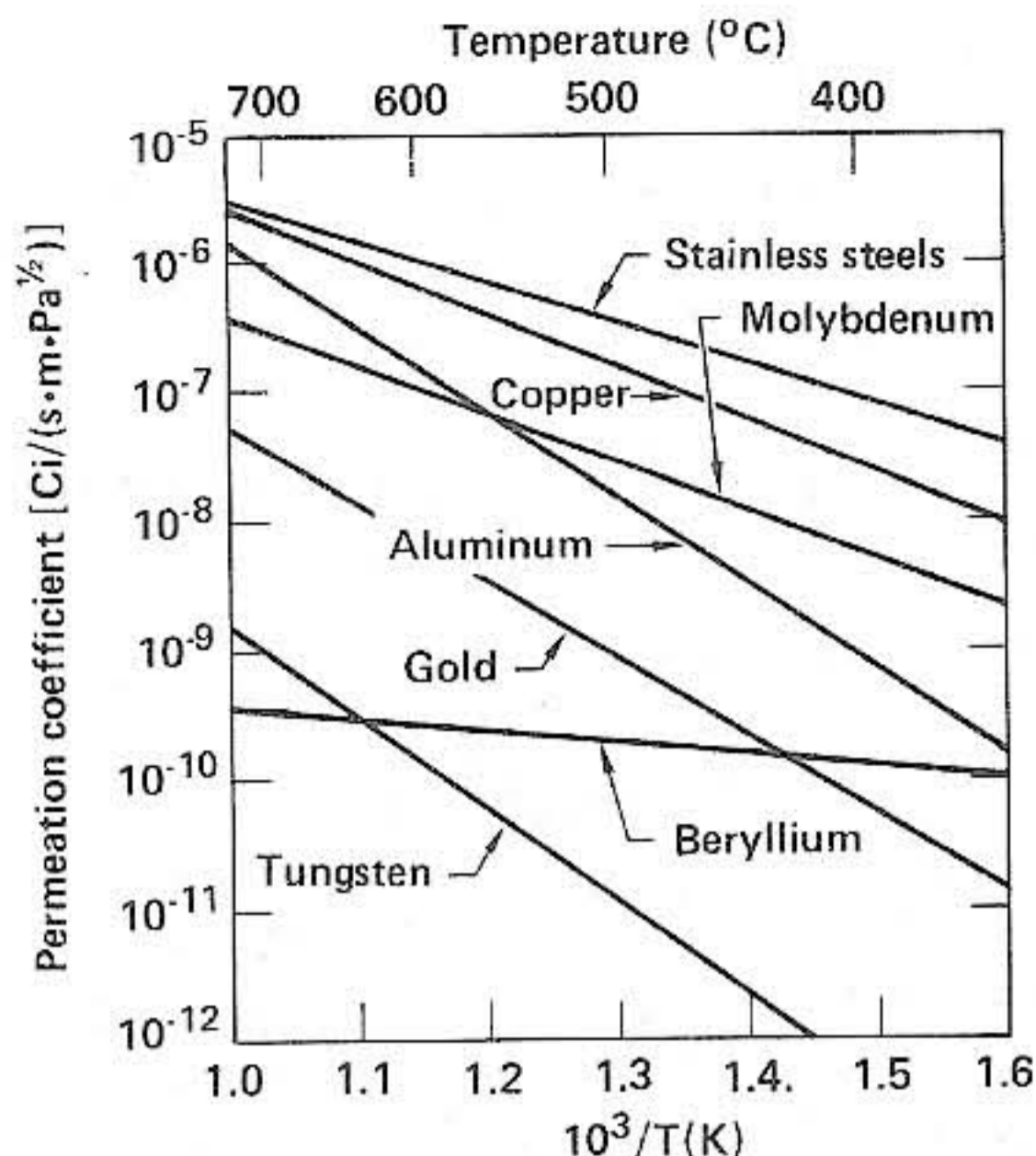


Figure 5. Permeability of various materials.

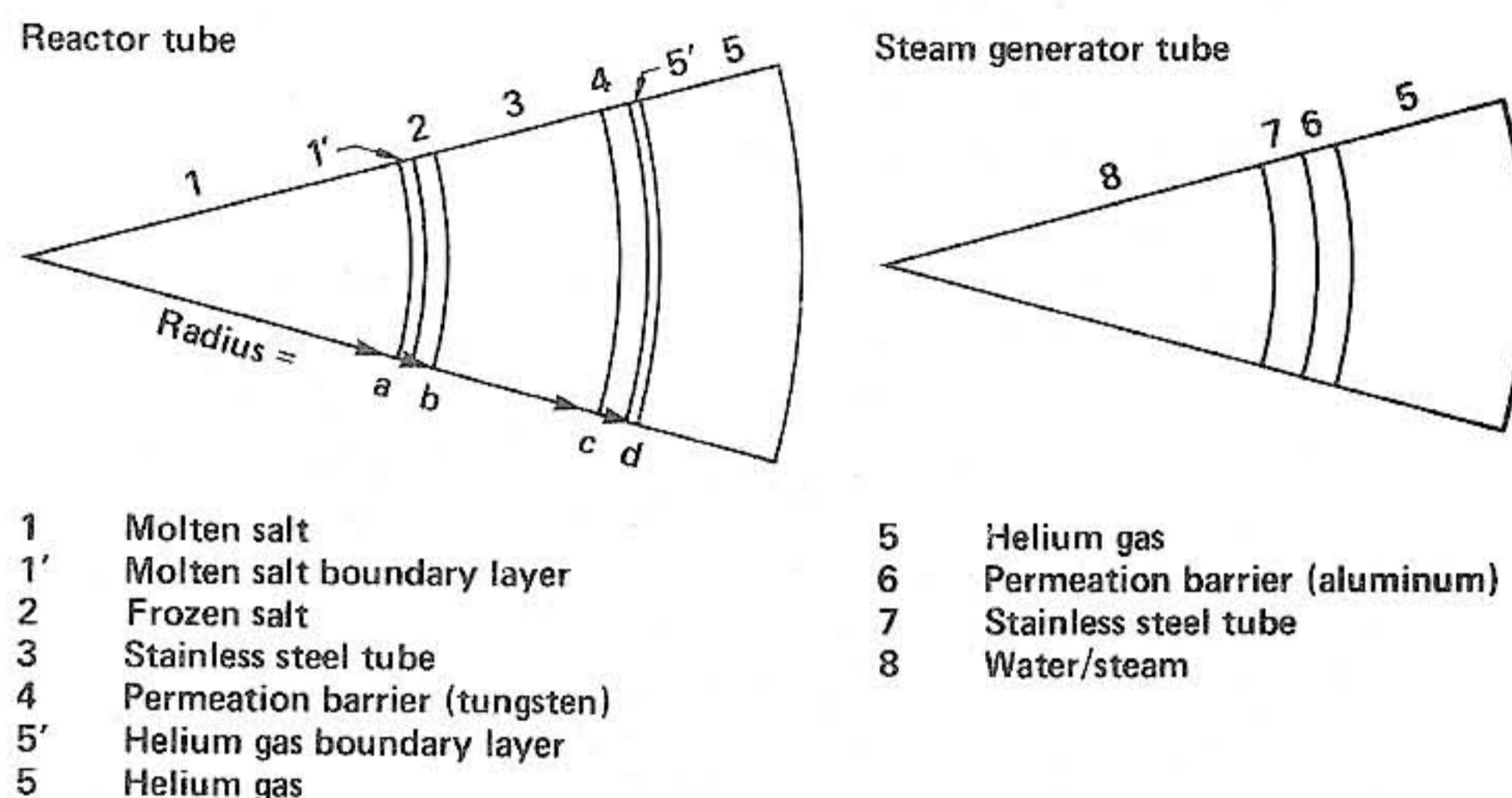


Figure 6. Permeation geometry and materials.

recovered as a hydride on a getter bed, with final cleanup accomplished by catalyzed oxidation and adsorption.

Finally, some definitive experimental work on the kinetics of tritium-gas conversion to tritiated water at low concentrations in helium is called for. Popular opinion has oscillated over the last decade from an initial optimism that thermodynamics would reduce the gas concentration to nil, to a current pessimism that predicts no gas conversion at all in the main helium loop. The critical experiments remain to be done, both with "clean" walls and particulate-free helium and in the presence of catalytic surfaces or other reaction promoters. The challenge is to demonstrate a method of drastically reducing tritium-gas partial pressure

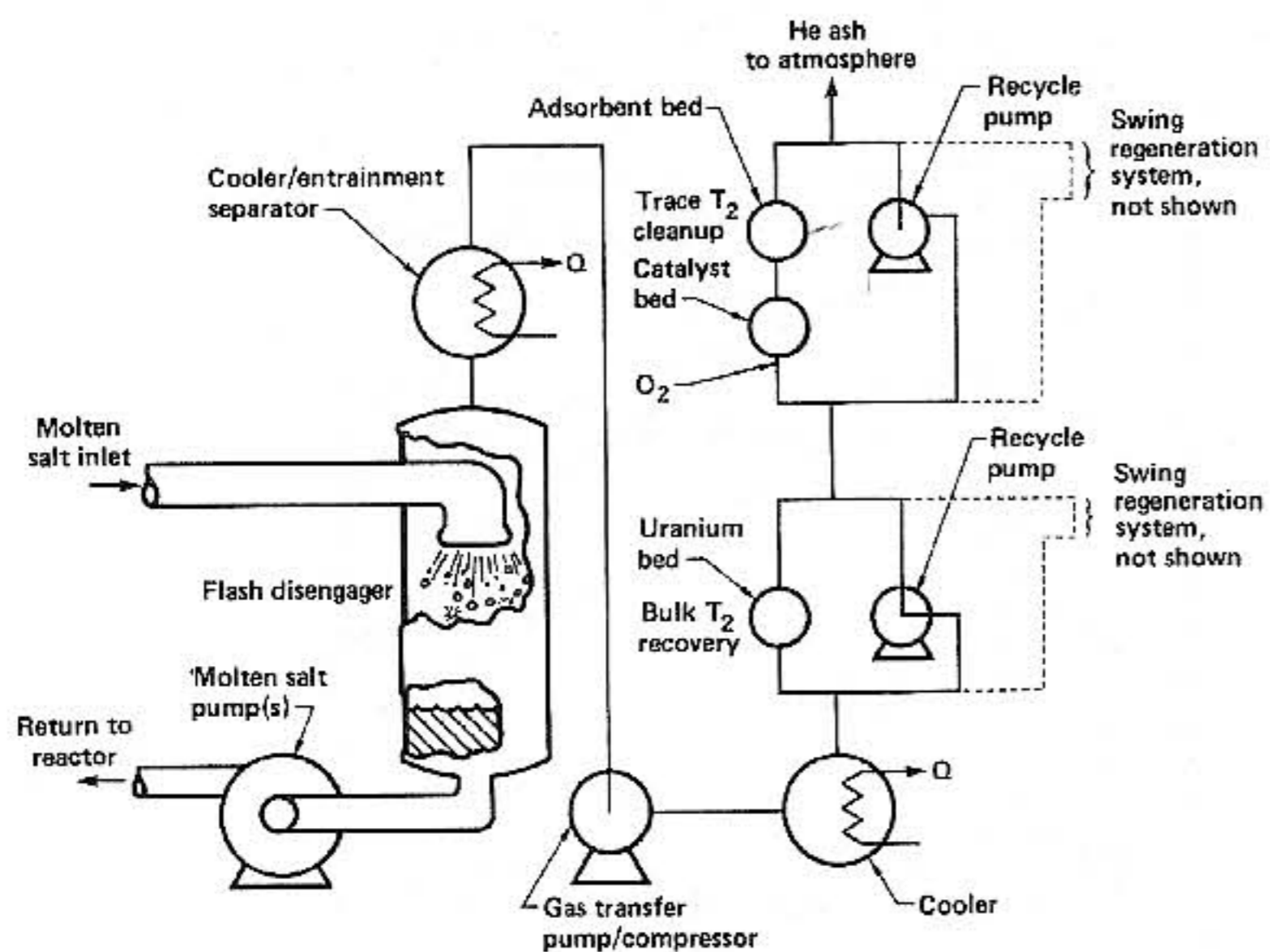


Figure 7. Tritium-recovery system flow sheet.

in the intermediate helium loop, and thus suppress permeation into the steam system.

BERYLLIUM-PEBBLE FABRICATION

Each beryllium pebble is a solid sphere of 1 cm nominal diameter and weighing 0.96 g. A nominal mass of 900 tonnes requires about 900 million pebbles for the initial inventory. See Table III for summary of beryllium parameters. For an assumed average pebble lifetime of two calendar years, the annual throughput of the hot beryllium fabrication plant will be 180×10^6 pebbles/yr or 180 tonnes/yr. However, with efficient recycle, the actual beryllium requirement associated with beryllium pebble remanufacture might be 1 to 10%, or 2 to 20 MT/yr. The selected pebble fabrication process involves the development of an automated line that will cold-press pebbles, vacuum-sinter them, hot forge them to full density, and vacuum-anneal them. To automate this process, a free flowing beryllium powder is required. Equipment for production of spherical beryllium is simple: mechanical presses and powder feeders to make cold pressed compacts, automated vacuum sintering furnaces for pressureless sintering, mechanical presses for hot sizing the sintered compacts, and an automated vacuum furnace for annealing the forged compact.⁹ For fabrication of pebbles too damaged to reinsert into the blanket after an irradiation period, we would vacuum melt the hot pebbles and use an automated atomization process (modeled after the Brush Wellman process) to first remanufacture the beryllium powder prior to the cold-press step. We do not exclude the possibility of removing the helium gas by vacuum heating, which would result in about 30% volumetric swelling. These enlarged pebbles might then be pressed back into size. The entire process will require provision for shielding and remote maintenance. In addition, hooding requirements as per OSHA limits ($2\mu\text{g}/\text{m}^3$) must

be maintained to limit airborne contamination. Given a conservative two-year beryllium lifetime,^{2,6} assuming an automated plant that operates 24 hr/da, 7 da/wk, and operates 85% of the time with no rejects, the production rate must be 110 balls/min. While this is a high production rate for beryllium parts, it is low for some powder metal industries (e.g., tantalum capacitor manufacturers produce thousands of parts/minute). The beryllium reprocessing line is estimated to lose 7 to 10% of the beryllium throughput, so a small feedstream is required. With a free-flowing powder, the losses might be reduced below 1%. An annual cost estimate for various assumptions was made (see Fig. 8).

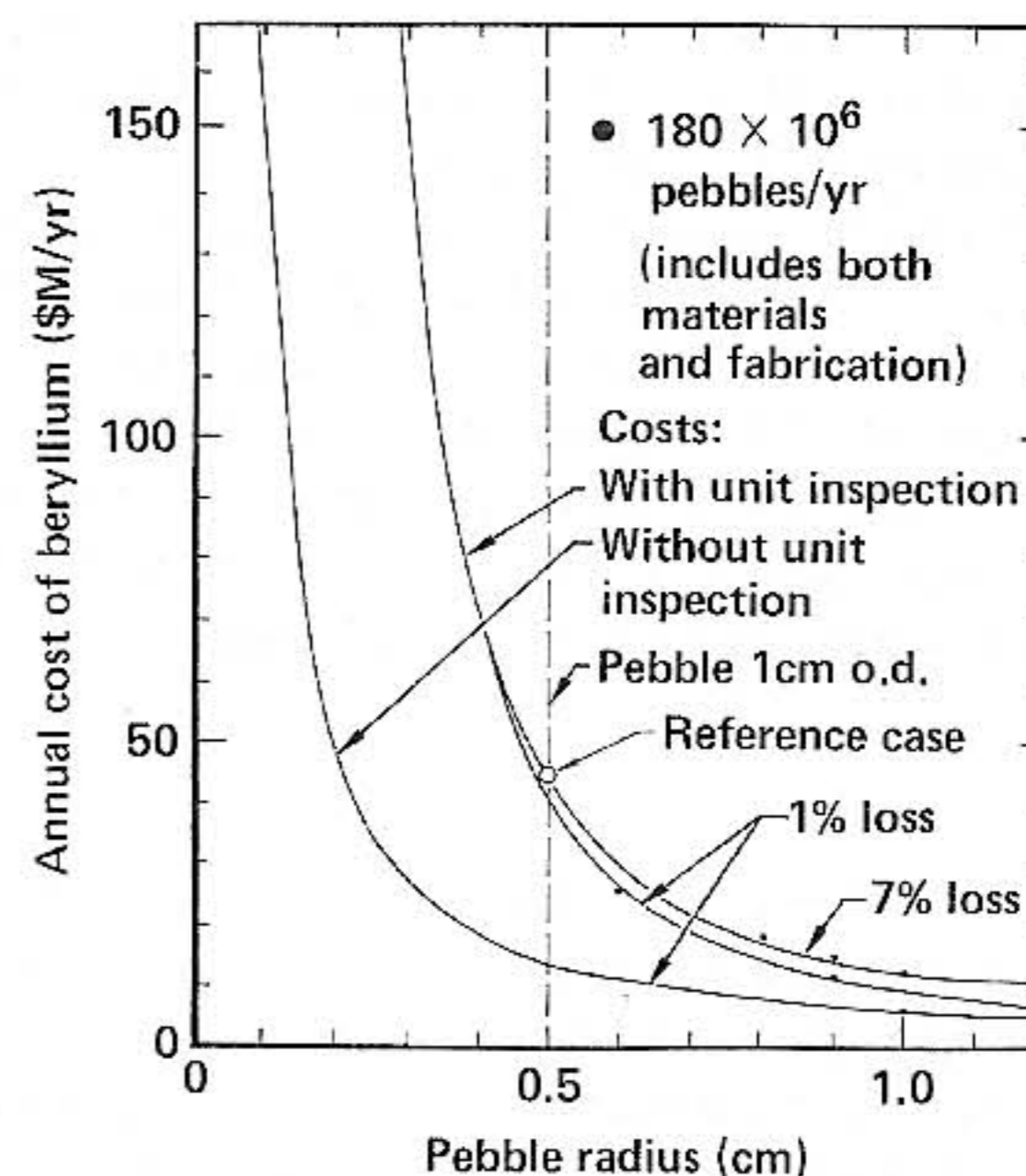


Fig. 8. Cost of beryllium pebbles.

HEAT TRANSPORT AND BALANCE OF PLANT

The blanket heat-transport system used helium as the coolant and transport medium. The major parameters of this system are listed in Table VI. The entire central cell is divided into 30 modules, and ten independent heat-transport loops remove heat from the blanket ($3,840 \text{ MW}_t$ total). Each loop serves three central cell modules and consists of a steam generator and a helium circulator. The steam generators envisioned here are based upon design features similar to those currently proposed for small high-temperature gas-cooled reactors (HTGR). The features of these factory-assembled steam generators include steel pressure vessels, helical coiled water/steam tubes, and once-through steam generation (Fig. 9). To keep the vessel-wall thickness low, the helium flow path may be arranged so that the cooled helium (293°C) from the economizer section exhaust is in contact with the wall. Helium piping in a helium-cooled system is a very important element because of its cost and its impact on building

Table VI. Heat-transport system of blanket.

Thermal power:	
Total blanket	3,840 MW _t
Added by helium circulator	110 MW _t
Blanket modules (no.)	30
Steam generators (no.)	10
Helium circulators (no.)	10
Thermal power of each	
steam generator	395 MW _t
Helium temperature:	
Blanket outlet/ steam-generator inlet	545°C
Blanket inlet/ helium-circulator outlet	300°C
Steam generator outlet/ helium-circulator inlet	293°C
Helium pressure:	
At circulator outlet	5,000 kPa
At circulator inlet	4,875 kPa
Helium pressure drop:	
Blanket	55 kPa
Steam generator	50 kPa
Piping	20 kPa
Total	125 kPa
Helium pumping power:	
Total	110 MW _e
Per circulator	11 MW _e (15,000-hp motor)
Helium flow rate:	
Total	3,000 kg/s
Per module	100 kg/s
Per circulator	300 kg/s
Steam-outlet temperature	510°C
Steam-outlet pressure	16.9 MPa
Feedwater-inlet temperature	193.3°C
Feedwater-inlet pressure	19.0 MPa
Steam flow rate per steam generator	159 kg/s

sizes (thermal-expansion requirement) and pumping power. Because of low density of helium, large size pipes are required. In this study the largest pipe size is limited to 1.5 m o.d. To keep the wall thickness low, internal insulation is used.

Steam power-cycle parameters, listed in Table VI, are primarily based upon helium temperature and state-of-the-art steam generator technology and its cost. Without the benefits of a cost tradeoff, we assume a reasonable temperature difference (ΔT) between helium and steam and a reasonable pinch point (lowest T) at the economizer/evaporator interface. Standard steam conditions (16.9 MPa, 510°C) consistent with the current turbine-generator practice are used. Tritium permeation through steam generator tubes is a major safety and economics concern.

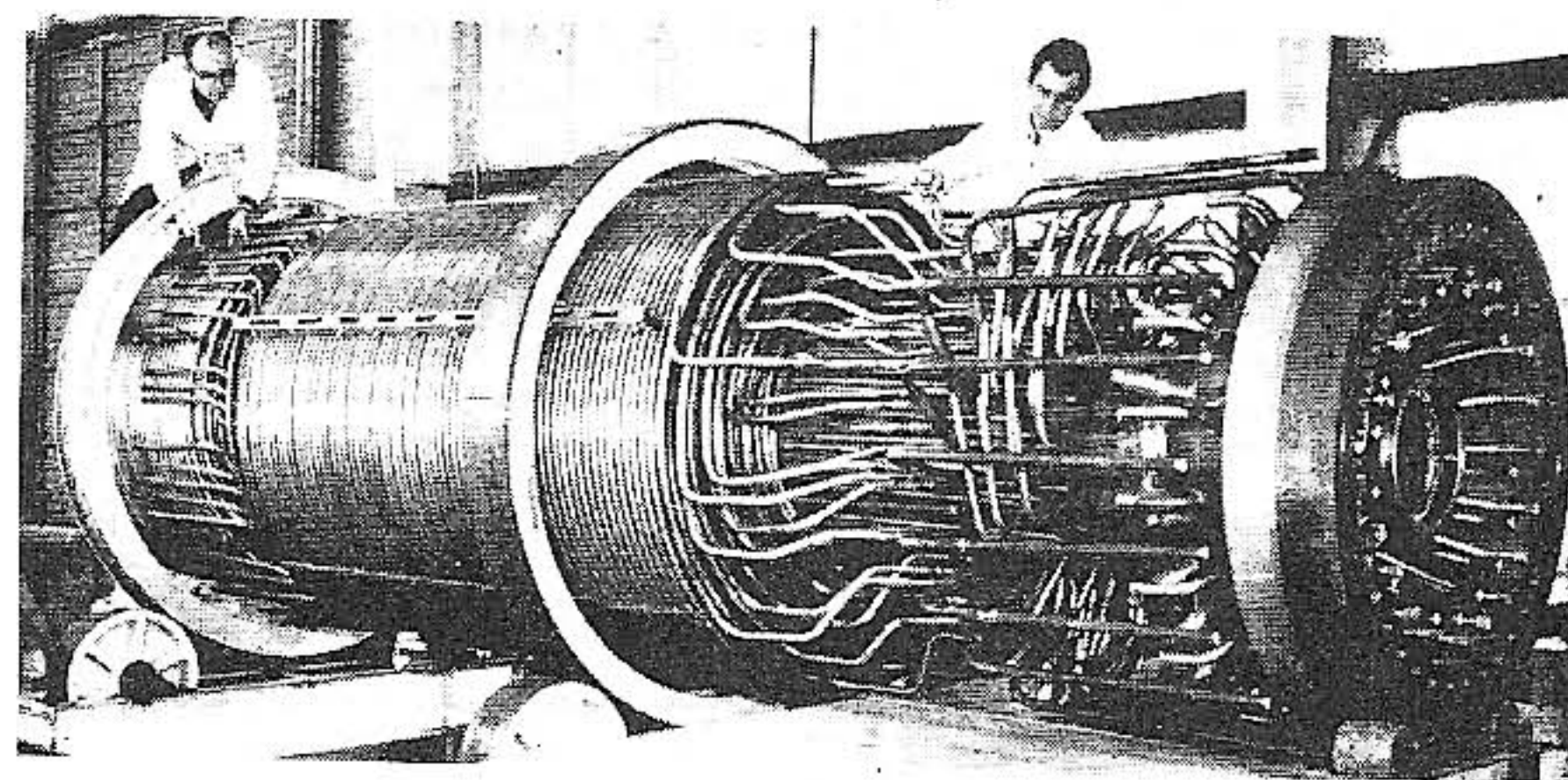
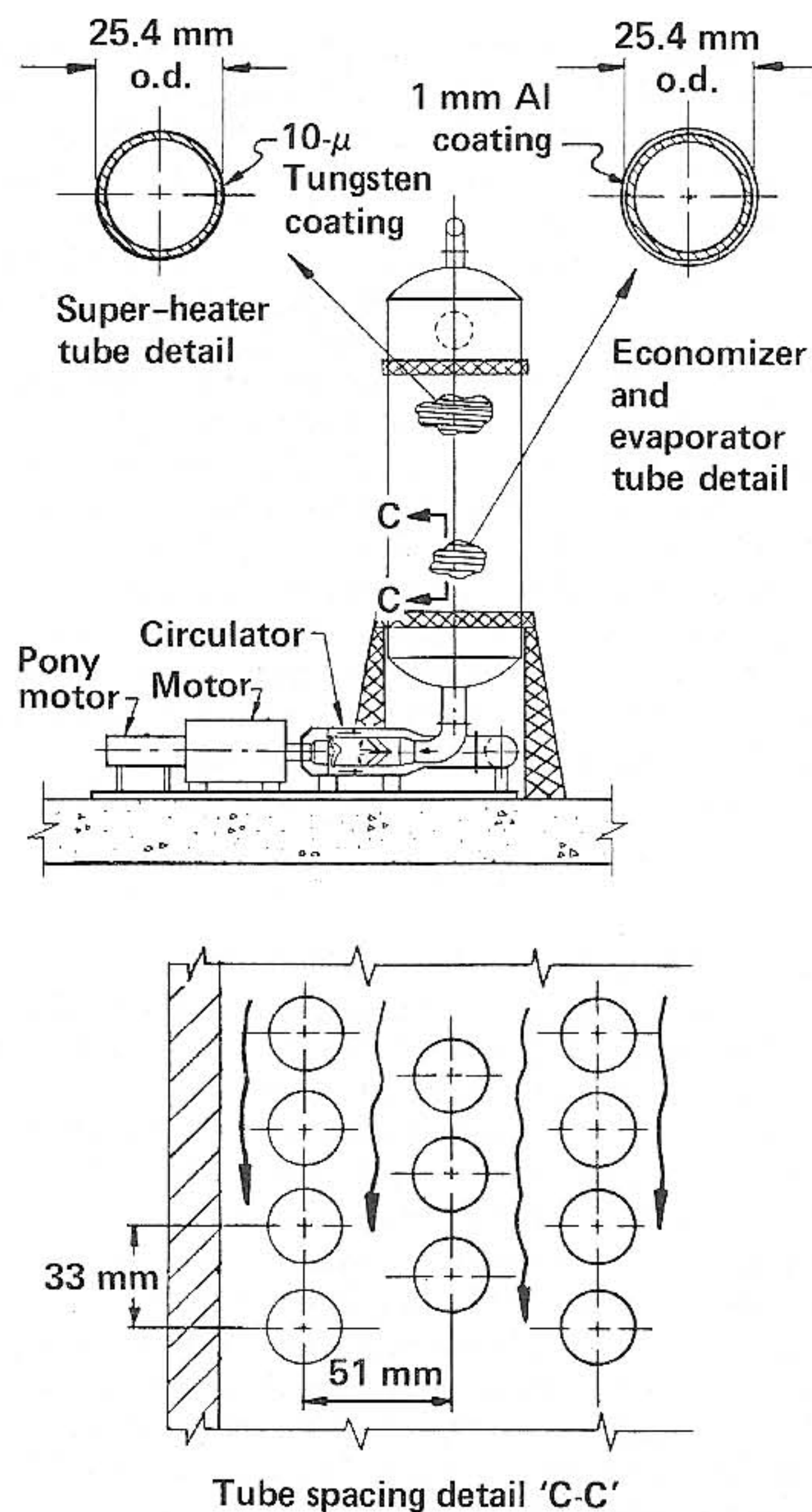


Fig. 9. Steam generator.

Two potential permeation barriers have been identified--tungsten and aluminum. When steam-generator tube surfaces are coated with either 10μm of tungsten or 1 mm of aluminum, the desired resistance to permeation is obtained. Aluminum coating on the inside surfaces is in contact with steam and may cause corrosion, while tungsten coating on the inside surfaces may exfoliate and cause damage to the turbine

blades. Tungsten coating on the outside surfaces also may exfoliate and cause damage to the helium circulator impeller. Aluminum coating on the outside surfaces is not suitable in the hottest (superheater) section of the steam generator, where aluminum in contact with the hot helium (545°) is too malleable. In general, aluminum with a much higher coefficient of thermal expansion than steel may tend to separate from the steel tubes, greatly reducing the heat-transfer coefficient. Use of alloyed steel (surface alloy of aluminum on steel base) may alleviate these problems; however, its permeation-resistive characteristics are not known. In addition to these potential problems, suitable methods of achieving the appropriate coating (uniform and reliable) are not available. Thus, much work, both theoretical and experimental, is needed to develop suitable permeation barriers.

INTEGRATED PERFORMANCE AND ECONOMICS

The overall performance and cost of the molten-salt fusion-breeder reactor are estimated and are combined with similar data for ^{233}U -burning LWR fission reactors to estimate the costs of electricity and bred fuel for a symbiotic electricity-generation system consisting of the fusion breeder, its LWR clients, and the associated fuel-cycle facilities. The performance parameters used to describe the molten-salt fusion breeder are given in Table I. The results of the economics analysis for two cases are given in Table VII: utility owned and government owned. The rationale for government ownership is the precedent set by government ownership of isotope-enrichment plants (diffusion and centrifuge plants); and since the fusion

Table VII. Economic analysis.

Cost analysis	Utility financed	Government financed
Total plant capital cost (\$M)	4867	4867
Cost ratio:		
Breeder/LWR	2.35	2.35
Total fixed-charge rate (%/yr)	15.05	9.05
Effective cost ratio		
Breeder/LWR	2.35	1.41
Year-one cost of elec. (mil/kW _e H)	50.8	46.9
Avg. PV cost of elec. (mil/kW _e H)	31.0	28.9
Year-one cost of ^{233}U (\$/g)	72.5	29.2
Avg. PV cost of ^{233}U (\$/g)	38.6	15.6

breeder in effect replaces an enrichment plant, it may be treated the same. The alternate may be for a consortium of utilities to own fusion breeders in order to have fuel-supply assurance. The difference in our analysis is the capital charge rate appropriate to the two cases, 9% and 15%. Another assumption is that electricity prices over the 30-year plant life are set by a fixed LWR cost and by a fuel cost whose U_3O_8 component rises 2%/yr in real terms because of resource depletion. The average present-value cost of fuel produced is \$16/g if government owned and \$39/g if utility owned. A price for ^{233}U of \$39/g is equivalent to about \$56/kg of U_3O_8 . The fusion-breeder cost estimate of \$4867M is 2.35 times an LWR cost and is based on a tandem mirror design prior to the more recent MARS⁵ (Mirror Advanced Reactor) design. Design improvements may lead to lower costs and substantially reduce the sales price of ^{233}U . The following conclusions result from modeling the potential economic performance of fusion breeders:

- The molten-salt fusion breeder could be economical at today's price of uranium (\$55/kg) regardless of ownership.
- With government ownership, the molten-salt fusion breeder could break even in the first year of operation and produce a multi-billion dollar benefit over its operating lifetime.
- Bigger benefits accrue for the molten-salt fusion-breeder operation at higher breeding and/or higher blanket-energy multiplication.
- A 20% breeding decrease is clearly tolerable.
- Some issues yet to be addressed include the potential impacts of lower LWR SWU costs, higher LWR fuel reprocessing costs, etc.

TECHNICAL ISSUES

Tritium Management. Permeation-barrier development and demonstration is the critical R & D need. Chemical-vapor deposition of tungsten on steel needs development work. Failure to develop effective barriers shifts the burden of processing tritium into the steam/water system. A successful barrier development effort could avoid both the need for water processing and the hazards that accompany tritiated-water systems.

Beryllium Feasibility. Beryllium-beryllium self-welding and beryllium-steel welding at the contact points are issues needing experimental investigation. The ability of the beryllium balls to stand up under neutron radiation is maximized by design (small size); however, irradiation data are needed. Beryllium swelling--leading to jamming, cracking, chipping, and flying

missiles that might damage circulators--needs investigation.

Material Compatibility. We predict the steel (type 316) will have a long lifetime in contact with the salt if it is kept in a reducing state. Corrosion tests with molten salt in a flowing loop would prove the predicted compatibility, including tests in fission reactors.

Neutron Economy. The breeding performance of the molten-salt design is very sensitive to parasitic absorption in structural material (principally iron). It is necessary to make better calculations of the nuclear performance and to emphasize designs that minimize structure material. Finally, experimental verification of tritium and fissile breeding should be carried out with a point 14-MeV neutron source.

CONCLUSION

The molten-salt fusion breeder design appears workable if the tritium containment and recovery of multiple distributed barriers and two-stream processing works as predicted. We have identified a number of R&D items that should be studied to increase our confidence level; however, the feasibility of making tritium barriers should be given the top R&D priority. The breeding is 6400 kg/year of ^{233}U at a cost of \$40/g for plants costing 2.35 times an LWR if utility owned or \$16/g if government owned and supports 14 LWR of equal thermal power. The design was carried out for a tandem mirror but should work equally well for a tokamak.

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