The Chemical Engineering Side of Nuclear Fusion Power

It is widely recognized that chemical engineering has important roles to play in the development of national and worldwide energy resources through optimal utilization of fossil fuel reserves. It is much less appreciated that there are crucial chemical engineering problems in the development of energy production from other sources. In particular, the successful development of nuclear fusion power generating systems will require the solution of many problems that are uniquely suited to chemical engineers.

This paper presents a brief overview of the fusion development program for magnetically confined fusion power reactors and an identification of the major technological problems remaining to be solved.

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SCOPE

A major purpose of this review is to alert chemical engineers to the technological obstacles that beset the path to the development of fusion power machines to the end that there will be increased participation in the development effort by those engineers and scientists best qualified to solve the problems. We first examine the potential importance of fusion power in the national and global energy economy and the relative urgency of its development. Next we present a perspective on the likely timetable for the development based in part on past history in the program and largely on current planning strategies. And finally, we describe the likely characteristics of fusion machines as a basis for identifying the major technological problems and the current state of the efforts to solve them.

CONCLUSIONS AND SIGNIFICANCE

This brief survey necessarily glosses over many problems in fusion development. It should be clear, however, that many of the technological problems are uniquely suited to chemical engineering. Because of the extreme conditions and inordinate constraints, the problems are difficult and challenging. Indeed, in a recent appraisal of current fusion research, Metz (1976) concludes that the problems appear overwhelming. Although his appraisal was focused on the plasma physics and materials problems, it would be easy to come to the same conclusion about some of the other problems discussed in this review.

On the other hand, it is true that fusion power is clean by comparison with fission power, it is inherently safer, and its ultimate fuel is available in limitless supply at essentially zero cost. Although its capital costs are necessarily high, it is hard to see that those for fission breeders or solar power for large scale power generation could ever be markedly less overall.

Given the difficulties of the problems remaining to be solved, fusion power must be regarded as a long range solution to our energy needs. But it is a solution, and possibly the best one. To realize practicable fusion power will require a major and continuing scientific and engineering commitment. The fact that we are well launched in that commitment is a happy exception to the notion that long range planning is impossible in democracies.

ENERGY PERSPECTIVE

Despite the large areas of ignorance and the enormous uncertainties which obscure attempts to forecast anything, it is inescapably clear that for the intermediate term, say to the year 2000, world population will continue to grow; the global per capita demand for energy, in particular electric power, will grow; and the available fossil fuels will decline in amount, while their real cost will continue to increase.

TABLE 1. ENERGY CONSUMPTION

	1970	2000	2050
World population	4*G*	7*G*	10*G*
kW(th)/capita	2	10	20
World energy consumption	0.24*Q*	2.*Q*	6.*Q*

$1Q = 10^{18}$ Btu $= 1.056 \times 10^{21}$ J
$1G = 10^9$
1970 U.S. energy consumption 0.07 Q/y, 10kW(th)/capita.

An unmistakable consequence is that technologically advanced societies must seek alternate fuel sources for generating power.

The only practicable alternatives to fossil fuels are nuclear sources, solar energy, and geothermal energy. The last two appear to be ruled out on economic grounds except for special local installations (Hottel, 1975; Rose, 1974). Conventional nuclear energy based on the fission of available high mass elements is a practicable alternative for the near term, but for the intermediate and long term, only the breeder fission reactor and the fusion reactor offer realistic possibilities. A full scale demonstration plant for liquid metal fast breeding fission power generation is now under construction, but sharp escalations in the projected costs and growing concerns about the hazards of plutonium have raised doubts about the practicability of this approach. On the other hand, fusion power generation has not even been shown to be feasible.

Given the uncertainties about any of the alternative sources of energy and our general inability to predict breakthroughs, it would be premature and certainly unwise to commit national resources in a major way to a single solution to the energy problem to the exclusion of other possible solutions. But it would be equally unwise to rule out arbitrarily any bizarre possibilities unless the contrary evidence is unmistakable.

ENERGY NEEDS AND AVAILABLE FUELS

An important driving force for the expansion of nuclear power generation capability is the declining availability of fossil fuels. Tables 1 and 2 present some data on current and projected demands for energy and on the likely recoverable resources. These data are adapted from a compilation derived by Hafele (1974) from a variety of sources.

For solar energy, the equivalent resource in Q per year is 1400, if we assume that the overall average solar energy deposition on the earth's surface is 90 W/m^2. Thus, if 1% of the solar energy incident on 1% of the earth's surface could be converted into useful energy, the contribution to the global energy supply would be 0.14 Q.

The implications of the foregoing tables and comments are first that coal and uranium will have to be relied on increasingly as primary energy sources for the near term and second, for the longer term, uranium or thorium in breeder reactors, and lithium and deuterium in fusion reactors, will have to be the principal fuel with possibly some assistance coming from solar sources.

HISTORICAL BACKGROUND

A brief account of the history of the development of fusion power will provide a perspective on some of the important problems and perhaps provide some clues as to the likely timetable for the emergence of the first commercial plant.

In the waning days of World War II, a classified theoretical effort under the code name Project Matterhorn was initiated at Princeton to explore the possibility of constructing a hydrogen bomb, that is, a nuclear weapon based on the fusion of low Z elements rather than the fission of unstable high Z elements. The theoretical feasibility of the hydrogen bomb was demonstrated fairly handily, and the project was transferred to Los Alamos for the actual construction of the bomb.

TABLE 2. FUEL RESOURCES (ENERGY CONTENT IN Q)

Sources	Known recoverable	Eventually recoverable
Coal	17.3	192
Crude oil	1.7	11.1
Natural gas	2.0	10.1
Tar sand oil	0.2	1.7
Shale oil	0.9	1.1
Uranium		
\$10/lb U_3O_8	0.7 (70)	30 (3 000)
\$100/lb		200-1 000 (20 000-100 000)
Lithium extractable from oceans)		7×10^5
Deuterium		3×10^8

NOTE: The figures for uranium are based on 1968 dollars. The numbers in parentheses are for breeder reactors operating at energy conversions of 3.5×10^{10} Btu/lb U_3O_8 (8.1×10^{13} J/kg U_3O_8). The numbers without parentheses are for light water reactors at energy conversions of 3.5×10^8 Btu/lb U_3O_8 (8.1×10^{11} J/kg U_3O_8). Thorium reserves are slightly greater than those of uranium.

Lyman Spitzer, an astrophysicist participating in Project Matterhorn, suggested that the fusion reactions involved in the hydrogen bomb might be carried out under controlled conditions such that a steady production of electric power would result. Based on this suggestion, the first conceptual design of a fusion power reactor system was undertaken. The resulting design report, called the Model D Stellarator Report (because it envisaged a development program involving three scaling models before the commercial prototype), was a surprisingly prescient document which anticipated most of the problems encountered so far (Spitzer et al., 1954).

In company with current thinking about the likely characteristics of fusion power machines, the Model D machine was visualized as a large, doughnut shaped vacuum bottle surrounded by a blanket filled with liquid lithium and liquid water, in turn surrounded by great magnet coils. Inside the reactor, fully ionized atoms of deuterium and tritium at temperatures in excess of 10^8 K, constrained by magnetic fields of 5 tesla or more to prevent impingement on the reactor walls, fuse to form helium and give off energy largely in the form of fast neutrons. The neutrons and their energies are captured in the blanket, and additional energy is produced there by the transmutation of lithium to tritium and more helium.

Essentially, all the energy picked up by the blanket is manifested as sensible heat in the liquid lithium and the moderator water. These streams are circulated through conventional boilers to generate steam for driving conventional turbogenerators to produce electricity.

The reactor is fed by injecting a steady stream of solid pellets of equimolar deuterium and tritium, the deuterium purchased from the outside, the tritium recovered continuously from the breeder lithium.

Since only a few percent of the feed reacts per pass through the reactor, a gas rate some twenty times the burnup rate must be circulated through the reactor. To assist this circulation, two large chambers (divertors) with appropriately shaped magnetic fields are inserted at two

TABLE 3. TYPICAL FUSION REACTIONS

$$D + T \rightarrow {}^4\mathrm{He} + n + 17.58 \text{ MeV } (94\,000 \text{ kWh/g}) \qquad (1)$$
$$D + D \rightarrow {}^3\mathrm{He} + n + 3.27 \qquad (2)$$
$$D + D \rightarrow T + p + 4.04 \qquad (3)$$
$$D + {}^3\mathrm{He} \rightarrow {}^4\mathrm{He} + p + 18.34 \qquad (4)$$
$$T + T \rightarrow {}^4\mathrm{He} + 2n + 11.32 \qquad (5)$$
$$\mathrm{H_2} + \tfrac{1}{2}\,\mathrm{O_2} \rightarrow \mathrm{H_2O} + 3 \times 10^{-6} \text{ MeV } (0.039 \text{ kWh/g}) \qquad (6)$$

points in the reactor to skim off and feed to vacuum pumps any ions that diffuse to the walls of the reactor. The circulated material goes through the whole temperature spectrum from plasma (that is, fully ionized gas) at 10^8 K to solid at a few kelvin.

The Model D report recognized that the plasma physics problems would be difficult and would have to be solved first. There was no first hand knowledge of the properties of plasmas at high temperatures; what little knowledge existed had largely been inferred from the behavior of stars like the sun. As a consequence, the initial activity in the fusion program was focused exclusively on plasma physics. Unfortunately, the problems proved to be so difficult that today, after more than two decades of continually expanding effort, the principal activity in the program is still concerned with plasma physics.

The first research facility in the field was the Princeton Plasma Physics Laboratory, and today it is the largest fusion facility in the United States and the only large one on a university campus. There are other large installations now at the Oak Ridge National Laboratory, at the Los Alamos Scientific Laboratory, at the Lawrence Livermore Laboratory, and in Europe and Japan. Many other laboratories in this country have growing activities in fusion.

At the outset all of the support for the program came from the Atomic Energy Commission, now the Energy Research and Development Administration (ERDA), but now an increasing volume of support is coming from the private utilities through the Electric Power Research Institute (EPRI) and other industrial sources, and work in laser fusion is supported to some extent by the Department of Defense.

PLASMA PHYSICS

It is beyond the scope of this review to describe in detail the problems of plasma physics, but some features of these problems dictate important characteristics of the power machines and hence many of the technological problems.

Although a very large number of fusion reactions are known, only a very few have a useful potential for power generation. Table 3, adapted from Post and Ribe (1974), lists the most attractive of these reactions together with the chemical reaction for the oxidation of hydrogen to provide a comparison of the energies involved.

In this table, n stands for neutron, p, proton, is the hydrogen isotope of mass 1, D, deuterium, the isotope of mass 2, and T, tritium, the isotype of mass 3. The reaction energies are given in millions of electron volts per event and in kilowatt hours per gram.

The fusion rate is given by

$$R = n_1 n_2 (\overline{\sigma v}) \qquad (7)$$

The bar indicates that the product of the cross section and ion velocity is averaged over the velocity distribution.

At high temperatures, the gas law applies and

$$n_i = P/kT \qquad (8)$$

Thus, from Equations 7 and 8 for any given maximum pressure, as might be dictated by magnetic and other constraints, the reaction rate will be proportional to $\overline{\sigma v}/T^2$.

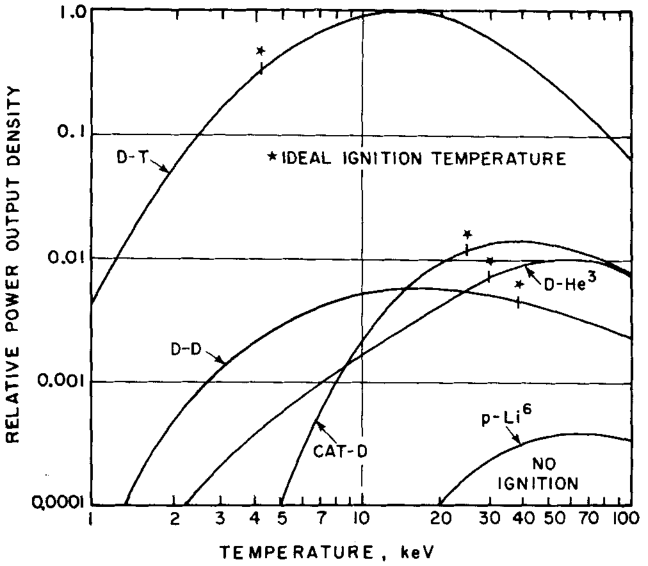

Fig. 1. Normalized output power density for various fusion fuels as a function of plasma temperature.

Figure 1, adapted from Mills (1972), shows a plot of this function, normalized as a relative power output density, vs. temperature for the fusion reactions listed in Table 3. The plot for the D-D reactions is an average for the two D-D reactions.

The temperature scale on the figure is given in kiloelectron volts which is a common mode of expression in plasma physics. It is actually an ion energy. The corresponding temperature in kelvin may be obtained from the fact that the Boltzman constant in eV/K is 0.86170×10^{-4}. Thus, 10 keV is approximately 10^8 K.

The vertical bars on the reaction rate curves are the estimated ignition points for the various reactions, that is, the lowest temperature for which the energy release to the charged particles is equal to the bremsstrahlung radiation losses (the principal mode of energy loss from the charged particles to bounding walls). From the standpoints of ignition temperature and rate of reaction, the D-T reaction is clearly superior to any of the other reactions. As a consequence, virtually all of the fusion research effort has been based on the assumption that this reaction would be the first one used for commercial power generation.

From available data on fusion reaction cross sections and energy release rates, it is possible to specify the plasma conditions requisite to the generation of power. The most commonly used specification of these conditions is the Lawson criterion, which states that the product of the ion density n_i and the reaction time τ for the D-T reaction at typical temperatures like 10^8 K must exceed 10^{14}, that is, $n_i\tau > 10^{14}$ s cm^{-3}. This result was obtained by Lawson (1957) by a straightforward energy balance on a system of n_i ions at T K for τ s assuming the energy release is converted to electricity at an efficiency of one third and used to preheat the fuel and sustain the radiation losses. A more realistic analysis by Mills (1971) is based on an energy balance around a steady state machine, assuming that only the energy release to the charged particles must support the heating of the fuel and the bremsstrahlung radiation losses. The result is a closed contour on the $n_i\tau$-T plane, with a minimum value at 10^8 K of 5×10^{14} s cm^{-3} for $n_i\tau$.

Since the temperature is essentially fixed by the ignition requirement, only the ion density and the confinement time can be traded off against each other. That is, confinement time may be short if the ion density is high enough, and vice versa.

Fusion can be accomplished at high densities by deposition of energy into a condensed mass of deuterium and tritium rapidly enough so that the ignition temperature is reached before the ions have a chance to fly apart. The principle here is inertial confinement, since it is the inertia of the fuel mass that provides the confinement. The hydrogen bomb employs this principle, using a fission bomb as the energy source.

The idea of using successions of hydrogen bomb explosions, for example, in salt caves, as a means of generating power has been studied under the code name Pacer (Hubbard et al., 1974), but the project is moribund. Current studies of inertially confined fusion systems for power generation involve the use of lasers or electron beams to drive the fusion.

Highly energetic, symmetrically deployed beams are focused on a small pellet of fusible fuel mixture in an effort to achieve conditions such that the pellet will implode and quickly reach ignition temperatures. However, the problems of achieving a reliable, essentially continuous power generator from a mini bomb machine seem enormously difficult.

The alternative to using inertial confinement is to employ forces that act at a distance and try to run under nearly steady state conditions rather than successions of shocks. The only forces that act a distance are gravitational forces, magnetic fields, and electrical fields. Gravitational fields achieve fusion, as the sun and other stars attest, but the scale is too large. Electrical fields cannot be devised to confine charged particles practicably. Magnetic fields, however, can readily contain charged particles and by adroit arrangement of field windings can produce virtually any desired shape of containment.

At attainable fields of the order of 10 tesla gauss, ion densities of 10^{14} can be achieved at temperatures of 10^8 K. Under these conditions, the Lawson criterion requires that the containment time be of the order of 0.5 s. Essentially all of the fusion research of the past 20 yr has been concerned with the problems of getting a gas heated to a temperature of 10^8 K at a density of 10^{14} cm^{-3} and holding it there for an appreciable fraction of a second. Two of the three conditions have been achieved simultaneously, but not all three. At the high temperatures obtained in the reactors, the gases become fully ionized, and their behavior in magnetic fields though subscribing in principle to Maxwellian physics has proved sufficiently bizarre to warrant calling them plasmas and considering them a fourth state of matter. A variety of devices has been proposed for achieving magnetic confinement of plasmas. Three types have been studied at length and are still under study, namely, mirrors, theta pinches, and tokamaks. Mirror machines, in their simplest articulation, are straight tubular reactors with high concentrations of magnetic fields at the ends of the tube to prevent escape of the ions to the outside by reflecting them back into the tube, hence the name mirror. Theta pinch machines comprise a toroidal reactor with an aspect ratio akin to that of a bicycle tire. They are operated in a strongly pulsed mode in that the reactor is charged with fuel, subjected to the magnetic field, and then pumped out, and the cycle is then repeated. Tokamaks (the word is a Russian acronym for a magnetically confined, toroidal reactor) have toroidal reactors with relative dimensions like those of a doughnut. They are quasi steady state devices with continuous feed and pump out.

Of the three machines, the tokamaks appear to offer the greatest promise of success, and the preponderance of current research on magnetically confined plasmas makes use of the tokamak principle which promotes the stability of the plasma by inducing a strong direct current in the plasma itself. Prior to the introduction of this principle, the approach to achieving fusion conditions in plasmas contained in closed systems like toroids had been seriously restricted by the onset of unstable gross motions in the plasma, leading to losses of charged particles to the reactor walls and resultant cooling.

Another problem with magnetically confined plasmas is the inevitable diffusion of the charged particles across the magnetic force lines to the walls. The simple solution to this problem is to make the minor radius of the reactor large enough so that the diffusion time is of the same order of magnitude as the desired confinement time. An unfortunate implication of this solution is that the minor radius of the reactor cannot be less than a meter. Thus, the demonstration of the feasibility of magnetically confined fusion requires a large and hence expensive experimental device.

For this demonstration, there is now under construction at the Princeton Plasma Physics Laboratory a tokamak fusion test reactor (TFTR). Scheduled for completion shortly after 1980, it will be the largest fusion test facility in the world and the first one to attempt the fusion of deuterium and tritium. The reactor vessel has an inner minor radius of 1.1 m and a major radius of 2.65 m. It will be operated in a pulsed mode in which a relatively cool plasma of tritium ions in the reactor is bombarded by a neutral beam of highly energetic deuterium atoms (Dawson et al., 1971). Although the device will not be a net power producer, it should convincingly demonstrate that fusion can be carried out under controlled conditions and that the basic plasma physics is sufficiently well understood to assure scaling-up to an experimental power reactor. The estimated price tag for this installation at the time of its design was $215 million.

FUSION DEVELOPMENT PROGRAM

Research and development for fusion power in the United States is funded almost entirely by the Division of Magnetic Fusion Energy (DMFE) of the Energy Research and Development Administration (ERDA). The current program plan of the Division for the development of fusion power by magnetic confinement, comprising a detailed four volume report, is succinctly summarized by Dean (1976). Five scenarios are presented ranging from a minimum funding level, which would result in a demonstration power plant in the far future, to a high funding level, which would provide a demonstration power plant before the turn of the century. Kintner (1976) has identified five major program milestones approximating the latter scenario. The first is to demonstrate reactorlike conditions in hydrogen plasmas by 1977 to 79. A principal instrument for this demonstration is the Princeton Large Torus (PLT) now involved in experiments at the Princeton Plasma Physics Laboratory.

The second milestone is the actual burning of deuterium and tritium with pulsed injection heating in the TFTR described above. This device is to be in operation by 1981.

The third and fourth milestones involve the successful operation of one or more experimental power reactors (EPR) generating several tens and more of megawatts of electric power. Based on the results of these steps and companion engineering studies, the fifth and final milestone is the operation of a demonstration power plant by 1995 to 1997 to ascertain overall reliability and economy. This unit might generate power at the 500 MW(e) or higher levels.

Fusion development efforts in other countries appear to be geared to similar scenarios with the exception that some

TABLE 4. SOME CHARACTERISTICS OF THE PRINCETON REFERENCE DESIGN FUSION POWER PLANT

Reactor	
Magnetic field	
a) At nominal center line of plasma ($R = 10.5$ m)	6 tesla
b) Maximum, at inside conductor	16 tesla
Plasma	
Confinement time	3.8 s
Volume	2 190 m^3
Percent burnup per pass	8.7%
Total reaction rate	1.22×10^{21} s^{-1}
Daily fuel consumption	
a) D_2	0.352 kg
b) T_2	0.527 kg
Tritium feed rate	4.545 kg/day
Fuel pellets	1 mm *DT* ice pellet with argon core to produce 4.8 atom % argon in plasma
Burn period per cycle	100 min
Shutdown period per cycle	3 min
Breeding blanket	
Thickness	74 cm
Daily fuel consumption	
a) Li^6	1.011 kg
b) Li^7	0.091 kg
c) Be^9	0.362 kg
Breeding fluid eutectic flibe	48 LiF-52 BeF_2
Volume (in blanket)	962 m^3
Mass (in blanket)	1.86×10^6 kg
Circulation rate	4.0×10^6 kg/hr
Net breeding ratio	1.10
Power	
Total nuclear power generated	5 305 MW
Net thermal power to steam generators	5 250 MW
Generator output	
Gross	2 407 MW
Available for sale	2 030 MW
Net thermal efficiency	38.3%
Tritium inventory	
Locations	
a) Primary fuel loop	0.50 kg
b) Salt processing loop: dissolved in salt	
recovery system	0.01 kg
c) Coolant helium:	0.10 kg
coolant	0.01 kg
recovery system	0.10 kg
d) Secondary containment, walls, etc.	0.03 kg
e) Reserve for startup	2.00 kg

intermediate steps may be skipped by taking advantage of the results of experiments here.

Although the proposed high level program in this country is not an emergency crash program, it is an aggressive one that will require a continuing escalation of budget from the current annual rate of about $200 million. Whether the actual program can adhere to the proposed timetable, of course, will depend on the political climate and on how well the plasma physics problems can be handled.

CONCEPTUAL DESIGNS

In the development of a novel engineering device like a fusion power reactor, it is desirable to preserve a wide spectrum of options in the early stages of the development to avoid cutting off alternatives which in the long run might prove to be advantageous. At the same time, however, it is desirable to undertake studies aimed at providing hardheaded engineering perspectives on designs achievable with existing, or nearly so, technology. These designs provide bench marks for appraising more speculative possibilities.

As a result of growing confidence in the likelihood of developing practicable ways of controlling fusion reactions for the generation of electric power, a number of laboratories have undertaken designs of hypothetical fusion power plants to provide useful references for gauging conceptual improvements. By exposing areas of uncertainty, these designs also provide guidance for experimental and development programs.

In addition to the pioneering Model D Stellarator design previously mentioned, there have been close to twenty other conceptual designs described in the literature. Most of these have been concerned with particular physical or technological problems and hence have not addressed the full spectrum of constraints that must be accommodated in a complete power generator. A summary of the features of nine different designs employing tokamak reactors is presented by Fraas (1975) in a detailed comparison of systems proposed by fusion groups at Oak Ridge, Brookhaven, Princeton, Wisconsin (Madison), Culham, Frascati, Julich, and Japan (Tokai). This study affords a good overview of the structural and materials problems involved in fusion power development. Davis and Kulcinski (1976) present a detailed parameter comparison of nineteen proposed designs for D-T tokamak fusion reactor including four near term (through 1985) test reactors, five mid term (through 1990) reactors, and ten long range (through 2020) power reactors. Their comparison takes account of plasma parameters and characteristics of blankets, shields, neutronics, tritium handling, materials, magnets, power cycles, resources, and economics.

Conceptual designs are described by Peterson et al. (1974) for a mirror machine and by Burnett et al. (1972) for a theta pinch machine.

Although the reactor schemes in these designs are quite disparate, much of the overall power systems have common features. For most of the systems the power generating equipment, comprising boilers and turbo compressors, is conventional. Furthermore, since most of the energy produced by the fusion reactions resides in energetic neutrons, all of the power systems must incorporate blankets to capture the neutrons. For D-T fuel cycle machines, these blankets serve essentially the same purposes regardless of the particular reactor scheme, but the designs proposed for them are almost as disparate as the reactors themselves.

The conceptual design we shall use as a basis for examining the full range of chemical engineering problems is the Princeton Reference Design (PRD), a hypothetical tokamak system requiring no new technology for its construction. We choose this design because it is the most detailed; we make no representation that it is the best design or that ultimate commercial plants will embody many of its features.

PRINCETON REFERENCE DESIGN (PRD)

The essential characteristics of the Princeton model are presented in Table 4, a cutaway drawing is shown in Figure 2, and a cross-sectional view is shown in Figure 3. A detailed description of this plant is presented in a 600 page design report prepared by some sixteen scientists and edited by Mills (1974).

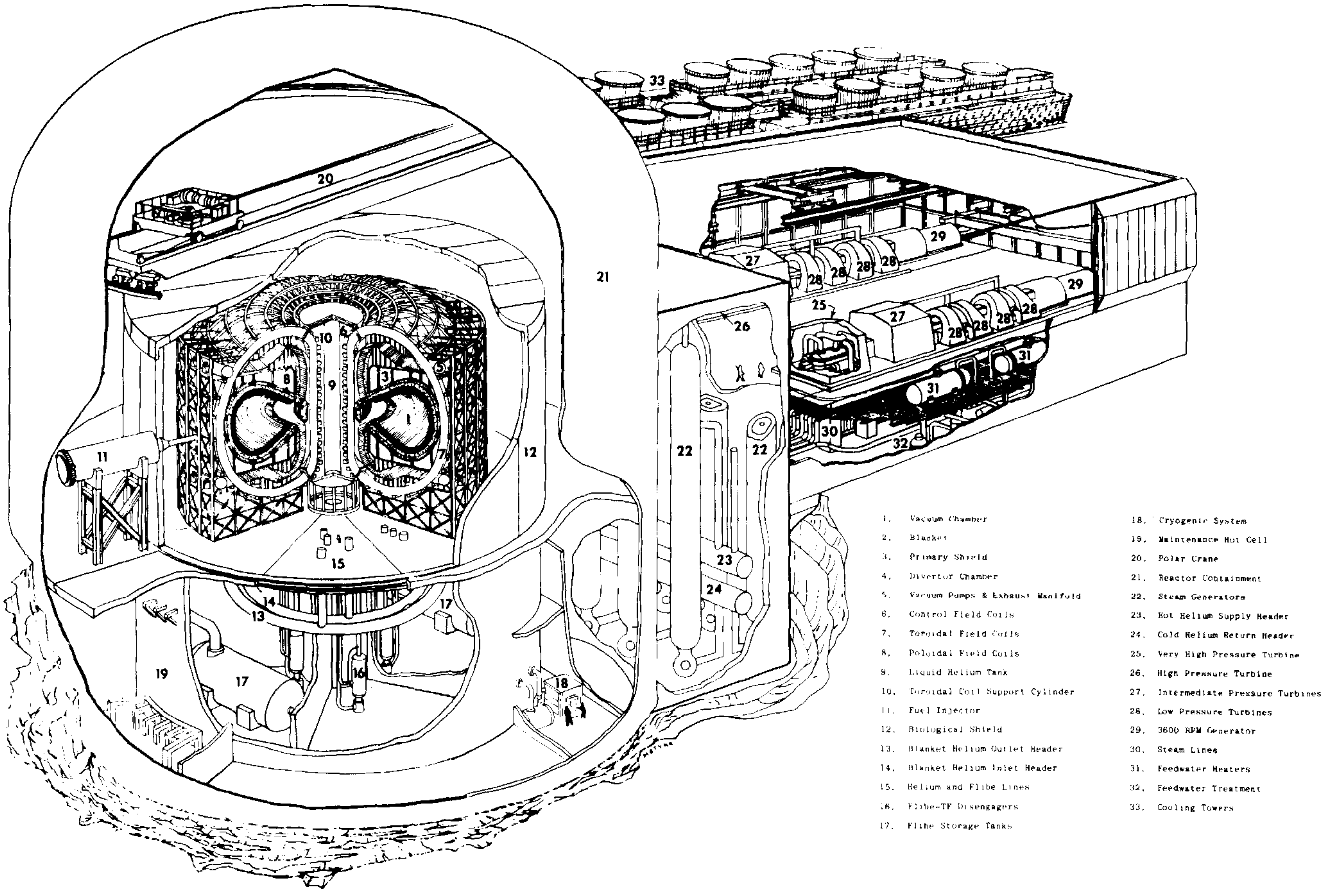

Fig. 2. Cutaway sketch of Princeton reference design.

In this machine, deuterium and tritium in the form of small frozen pellets are injected continuously into a tokamak reactor. Nearly 10% of the fuel ions fuse at the conditions prevailing in the reactor, and the resulting fast neutrons are captured in a blanket consisting of a molten mixture of lithium fluoride and beryllium fluoride, cooling tubes containing helium gas as coolant, and structural members.

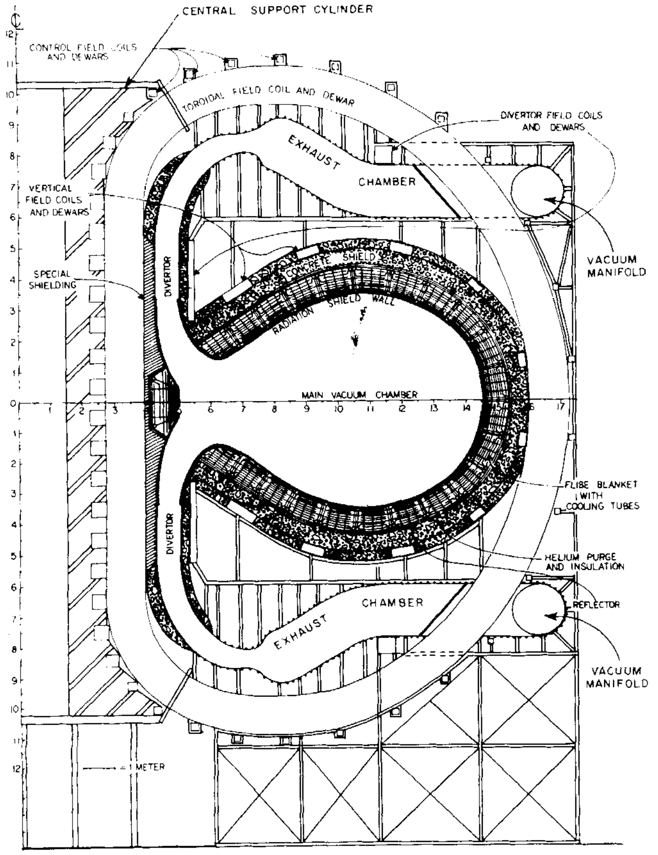

Fig. 3. Reactor cross section (one half).

The principal fusion reactions are

$$D + T \rightarrow n + He + 17.58 \text{ MeV} \qquad (9)$$

and the principal capture and neutron multiplication reactions are

$$Li^7 + n \rightarrow T + He + n - 2.47 \text{ MeV} \qquad (10)$$

$$Li^6 + n \rightarrow T + He + 4.78 \text{ MeV} \qquad (11)$$

$$Be^9 + n \rightarrow 2He + 2n - 1.58 \text{ MeV} \qquad (12)$$

Since the transmutation of lithium-6 as given by Equation (11) is the preponderant fusion reaction in the blanket, the nuclear energy release from reactions in the blanket comprises a large fraction, nearly one fifth, of the total energy generation in the plant.

In the PRD all of the energy release is collected as sensible heat in circulating helium gas compressed to 50 atm (5 MN/m^2). The breeder salt, comprising a 48 to 52 molar mixture of lithium fluoride and beryllium fluoride (flibe) is also circulated continuously, but it is not cooled. Its circulation rate is fixed to permit desorption of the tritium and helium gases formed from the breeding reactions. This rate suffices to prevent excessive temperature rises in the salt while it is in the blanket.

The coolant helium flows through conventional heat exchangers, which generate high pressure steam to drive turbogenerators for electric power production.

With the above brief description as a base, let us look in some detail at the various parts of the plant where there are important chemical engineering problems.

PRIMARY FUEL LOOP

An overall perspective on the fuel flows is presented in Figure 4, which shows the three circulating loops principally involved in fuel handling. The coolant helium loop is included because there is a finite transport of tritium into the helium at many points in the cooling circuit by virtue of the permeation of metal walls at high temperatures. The elements of the primary fuel loop itself are shown in greater detail in Figure 5.

Since this loop contains the reactor, the critical problems here are those of plasma physics. Almost as critical, however, is the problem of feeding the fuel into the reactor.

For quasi steady state machines like the PRD, the fuel must be fed continuously either as a stream of pellets or droplets or as an energetic beam of neutral atoms. As mentioned earlier, the TFTR will employ neutral beam injection, which also provides a powerful means of heating the plasma. Indeed, neutral beam injection can effect fusion reactions below ignition conditions.

The PRD uses solid pellet injection (Gralnick, 1974), wherein frozen droplets approximately 1 mm in diameter are accelerated electrostatically to velocities in excess of 10^4 m/s and fed through some forty injectors distributed about the reactor at a total rate of about 300 pellets/s. High velocities are necessary to ensure that the pellets penetrate deeply into the plasma before they evaporate and become immobilized by the magnetic field as they ionize. C. D. Hendricks (Gralnick, 1974) has demonstrated that solid droplets can be accelerated to these high velocities, but there are large areas of ignorance in this problem, and relatively little effort is focused on it.

In the reactor the ions circulate for over 3 s before diffusing out toward the walls, where they are continuously skimmed off by the divertors. In that time nearly 10% of the deuterium and tritium fuel ions fuse. The ions entering the divertors are cooled by contact with cold particles and with cooled metal surfaces. They recombine to molecules and after further cooling are pumped out by mercury diffusion pumps.

Bombardment of the divertor surfaces and to a lesser extent the reactor walls by energetic ions results in some ion implantation in the metal. Since the temperatures of most of these metal surfaces exceeds 300°C, the hydrogenic species (mostly deuterium and tritium, but also small amounts of protium) will tend to diffuse through the metal walls and enter the coolant helium. The more energetic ions will also tend to sputter off metal atoms, which in the reactor can result in exessive cooling and possibly quenching of the fusion reactions.

More serious than the effects of the ions, however, are the effects of the neutron flux through the first wall of the reactor. Indeed, many in the fusion scientific community regard the first wall problem as second only to the plasma physics in difficulty and challenge (Kulcinski, 1976).

Although the metals suggested as candidates for the first walls of fusion reactors are relatively transparent to neutrons, the fluxes in fusion power reactors are so high ($10^{19}/m^2$ s for the PRD) that every atom in the metal lattice of the first wall is estimated to undergo ten to thirty displacements per year. Depending on the thickness and nature of the first wall, as much as 5% of the total energy production of the machine can be deposited there with the result that special cooling and thermal stress problems must be accommodated.

Since the PRD is designed to use available technology, the principal material of construction proposed for the vacuum wall of the reactor and for piping and structural elements within the blanket is the Wiggins alloy Nimonic PE-16, a high nickel, austenitic alloy. Other materials suggested for this exacting duty include other stainless steels and refractory metals like molybdenum, vanadium, and niobium and related alloys.

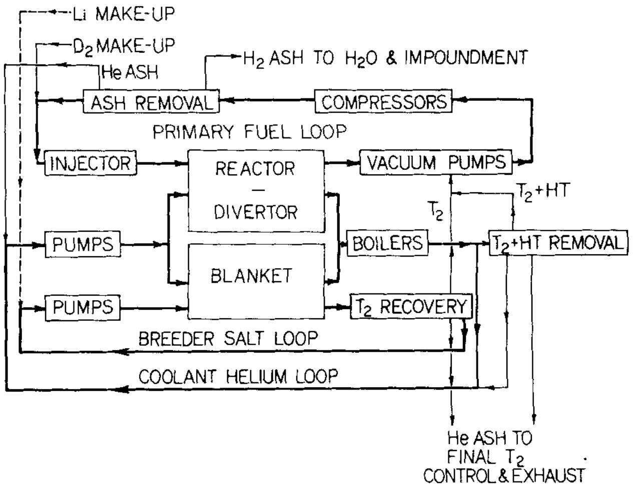

Fig. 4. Principal fuel flows.

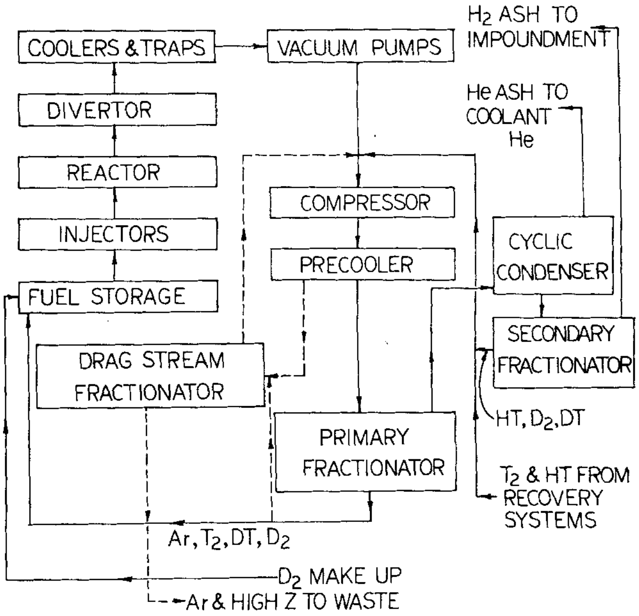

Fig. 5. Primary fuel loop.

For systems using liquid lithium as the breeding medium, the refractory alloy of niobium containing 1% zirconium, Nb-1 Zr, has many attractive properties, not the least of which is that it can withstand temperatures in excess of 1 000°C (Fraas, 1973).

It is clear that the materials problems are particularly severe within the reactor-blanket region. Intense radiation, steep thermal gradients, corrosive attack by breeder fluids, and the need for maximum temperatures to achieve high thermodynamic efficiency all combine to constrain narrowly the range of practicable choices. Only careful and costly experimentation can identify the ultimate materials.

A detailed accounting of current researches is given in the first two volumes of the proceedings of the International Conference on Radiation Effects and Tritium Technology for Fusion Reactors held in Gatlinburg, Tennessee in October 1975 (Watson and Wiffen, 1976).

PRIMARY FUEL PROCESSING

The gases exhausting from the divertors via the vacuum pumps contain mostly deuterium and tritium plus a few percent each of helium and argon and something less than a percent of ordinary hydrogen (protium). Argon is added to the PRD feed to prevent excessive escalation of electron temperatures in the reactor. Its presence slightly complicates the processing of the primary fuel loop gases.

Since the reactor feed must be injected under cryogenic conditions, the simplest strategy for purifying the recycle gas is to make use of cryogenic fractional distillation. By using small packed columns with separating powers equivalent to fifty theoretical plates and operating them at essentially total reflux, it is possible to eliminate the helium and protium ash. Some of the protium is present initially as TH, and consequently catalyzed equilibration is necessary to ensure the separation of the protium from the tritium. The scale of all operations here is small, but there are no basic difficulties since the volatilities of the various isotopic compounds are well spread out. Experimental studies at the Mound Laboratory confirm the practicability of cryogenic fractional distillation (Wilkes, 1975). A typical arrangement of process elements is shown in Figure 5.

An alternative method that has been propsed for processing the primary fuel loop is multistage permeation. This method easily separates the helium and argon from the hydrogen isotopes, and theoretically it should separate the D, T, and H since the permeation proceeds atomically and the respective rates should go inversely with the atomic mass. The latter separation has not been demonstrated practicably, and it is possible that surface phenomena participating in the overall transport may confound the separability provided by the metal membrane alone.

BLANKET CHARACTERISTICS

In a typical fusion power machine the blanket surrounding the reactor plays a number of important roles. First, it must be a tritium breeder, capturing enough of the neutrons leaving the reactor to ensure a favorable breeding ratio for tritium; second, it must be a shield for the magnets, protecting them from neutrons and radiations which could cause damage or impose excessive heating loads on the cryogenic system which cools the magnets; and third, it must provide for the transfer to the power generating elements of the bulk of the energy produced by the fusion reactions. For inherently pulsed or cyclic devices like tokamaks, the blanket with its necessarily large thermal capacitance should also serve to provide most of the thermal ballast in the system so that the net energy flow to the power grid is at constant rate despite the fluctuating energy flow from the reactor.

CHOICE OF BLANKET FLUIDS

In early conceptual designs of fusion power reactors, there was a common tendency to choose a single fluid to fulfill in large measure all the functions of the blanket. For example, the earliest conceptual design, the Model D Stellarator, proposed the use of liquid lithium metal as breeder, coolant, and principal shielding material. Subsequently, in part stimulated by a concern for the practicability of pumping an electric conductor through magnetic fields, a number of blanket schemes were proposed with molten salt as the principal multipurpose blanket fluid. One of the earliest of these schemes (Johnson, 1957) suggested the use of molten lithium nitrate or nitrite or mixtures thereof and slurries of lithium oxide crystals in these molten salts to increase the lithium density in the blanket. The cheap alkali nitrates and nitrites (those of sodium and potassium) have long had practical use as heat transfer media in the chemical process industries. In this use they have proved chemically and thermally stable up to moderately high temperatures and noncorrosive toward mild steels. As blanket fluids, the corresponding lithium salts would have the advantage that any tritium bred in them would be tightly bound up as water and readily recoverable as such.

Unfortunately, the moderately high temperatures achievable with the nitrates and nitrites are low (500°C maximum) by power plant standards, and because the anions contain covalent linkages, it is likely that these salts would be unstable toward neutronic bombardment. Consequently, virtually all subsequent proposals of molten salts for blanket fluids involve mixtures of lithium fluoride and beryllium fluoride (Homeyer, 1965; Grimes and Cantor, 1972).

The choice of blanket fluid is severely constrained. Stability in high neutron fluxes and other radiation and at high temperatures, freedom from adverse neutron capture reactions, and inertness toward bounding walls and structural materials are required of all fluids. In addition, for breeding fluids high lithium densities and easy tritium recovery are necessary. For coolants, the thermal properties and flow properties (including electrical conductivity since flows across magnetic lines are unavoidable) must be favorable.

If the plant is constructed using current fabrication technology, the principal material of construction for the blanket will most probably have to be austenitic steel alloys. For these materials it is likely that higher average operating temperatures can be maintained without excessive corrosive attack on metal walls if molten fluorides are employed as the lithium bearing breeder rather than liquid lithium. This temperature advantage, which exceeds 100°C, with corresponding advantage in thermodynamic efficiency, coupled with the ease of tritium recovery from molten salts, led to the choice of molten fluorides as the breeding medium for the PRD. An additional factor was the wealth of successful experience in the molten salt breeder experiment at Oak Ridge in handling molten fluorides under conditions of high temperature and moderately intense radiation (Weinberg, 1957). A countervailing factor is the poorer tritium breeding capability of molten salts in comparison with pure lithium.

It should be emphasized that the choice of molten fluorides for the principal blanket fluid in the PRD was dictated by the requirement that the design avoid incorporating features involving new technology. Most other conceptual designs employ lithium metal or solid lithium compounds or alloys as the breeding medium.

BREEDER BLANKET

Although the neutronics of breeding tritium in fusion reactors lies outside the scope of this article, it will be helpful to examine the constraints imposed by the need to maintain a neutron balance. In principle, one neutron born in the plasma can produce one triton by absorption in a lithium-6 atom, and something better than that by absorption in lithium-7. The capture cross section for the former is very high for thermal neutrons, that is, low energy neutrons, and for the latter it is relatively high for fast neutrons. Numerous calculations have been performed for wide varieties of breeding systems to appraise the practicability of different choices of breeding media, moderators (to slow down the fast neutrons), structural materials, and cooling agents.

Two useful parameters for characterizing the breeding efficiency of a particular reactor system are the breeding ratio, namely, the number of tritons produced in the blanket per fusion neutron produced in the plasma, and the doubling time, which is the time required to produce a net amount of tritium over the fuel requirements of the plant equal to the total normal tritium inventory in the plant. For typical hypothetical designs, these parameters range from 1.03 to 1.5 for the former and a few months for the latter.

For systems using natural lithium as both breeder and coolant, it is possible to achieve satisfactory breeding ratios without additional neutron multipliers beyond the lithium-7. With another material as coolant, for example, helium, the volume of the blanket occupied by the cooling tubes and coolant results in lower breeding ratios, and the addition of beryllium is helpful in restoring the ratio.

Systems involving molten salts typically require neutron multipliers for acceptable breeding ratios (Price, 1973). In the case of fluorides, the presence of beryllium fluoride is also necessary to lower the melting point of the salt to workable levels. The melting point of lithium fluoride is 848°C, but a 50-50 mixture of lithium fluoride beryllium fluoride melts below 380°C.

Although other salts have been considered as possible breeding media, only the fluorides have the ionic binding necessary for stability in a neutron radiation field together with practicable melting points and freedom from parasitic neutron capture reactions. Imamura (1977), for example, has shown that the chlorides are much less attractive. In principle lead, which has favorable neutron multiplication characteristics, could be substituted for beryllium. The major argument for making this substitution, however, is that beryllium is a limited resource, but that argument is probably invalid. Beryllium is a low Z element and is likely to be relatively abundant in the earth's crust. The fact that the known reserves are low may merely reflect the absence of a pressing need to find more.

Studies of solid breeding media, conducted largely at the Brookhaven National Laboratory and in Japan, have shown that adequate breeding can be obtained with a variety of materials including lithium aluminum and lithium lead alloys, Li_2O and lithium aluminates and silicates (Powell et al., 1973). These systems necessarily require that the blanket be cooled by something like helium.

BLANKET COOLING

Except for the relatively small amounts of energy deposited in the first wall and the divertors, all of the energy generated in tokamak type of reactor systems winds up in the blanket. The principal coolants proposed for removing this energy are lithium liquid, molten salts (lithium beryllium fluoride in particular), and helium.

Liquid lithium circulating in a multiplicity of small loops and giving up its energy to boiling potassium has been proposed for the Oak Ridge design of Fraas (1973). The magnetohydrodynamic losses in transporting the lithium across the magnetic field lines have been shown to comprise less than 1% of the power output of the machine based on experimental studies of a one sixth scale test loop (Young et al., 1975). Recent calculations for the use of lithium as coolant in UWMAK machines (Conn, 1976) show that the pumping power requirements will approximate 1.5% of the gross plant output. The maintenance of low pumping costs requires adroit engineering to ensure that the flow paths of the lithium minimize the crossing of the magnetic flux lines.

Lithium may be used as coolant at operating temperatures as high as 1 000°C provided refractory metals like Nb-1 Zr are used. If commercially available materials like stainless steel must be used, the maximum temperature is something like 500°C.

In principle it might be expected that molten salts could be useful coolants for fusion reactor blankets. They have relatively high heat capacities and are reasonably mobile in magnetic fields, and because of this mobility, their somewhat lower thermal conductivities (in comparison with metals) should still admit of a practicable heat transfer capability. Actually they are marginal heat transfer agents. Since their thermal conductivities are relatively poor, molten salts must be pumped at sufficient velocities to ensure adequate heat transfer by convective mechanisms. In the intense magnetic fields prevailing in the blanket of fusion reactors, 5 to 10 tesla gauss, high velocities may produce large electric potential differences between the salt and the walls wherever the salt flows normal to the magnetic field lines (Grimes and Cantor, 1972). The effect of these potential differences would be to promote electrolytic interaction between the salt and metal with a sharp enhancement of local corrosion of the metal walls. Thus it is likely that molten salts in fusion reactor blankets should only be moved in laminar flow through the blanket, in which flow regime they have little utility as coolants.

A further and much more stringent limitation on the usefulness of molten salts as coolants is that they may permit the transfer of tritium from the reactor system to the steam cycle. This transfer could come about if the coolant salt were circulated from the blanket directly through the steam generators of the power plant, and if the molten salts permitted an appreciable tritium pressure at the walls of the steam generator tubes.

Tritium can exist in molten fluorides as either molecular tritium or tritium fluoride depending on the oxidation state of the molten salt. As molecular tritium it will migrate to the bounding metal walls, and at the temperatures that prevail in the steam boilers it will dissociate at the tube wall surfaces and diffuse through the walls ultimately winding up in the steam, contaminating the steam, and precluding any useful recovery of the tritium. As tritium fluoride it will also migrate to the tube walls and attack metals there according to reactions like

$$2\,TF + M \rightarrow T_2 + MF_2 \tag{13}$$

Actually, the tritium formed in this reaction initially will be in nascent atomic form, and its tendency to permeate the metal wall might be greater than would be predicted from the equilibrium molecular tritium concentration.

With liquid lithium as coolant, the possibility of tritium permeation into the steam cycle would be much less than for molten fluorides for the same tritium inventory because the tritium pressures are much lower. On the other hand, the hazard resulting from leaks in the steam generators that would mix lithium and water at high temperatures might be objectionably great.

The concern about tritium diffusion or leakage into the steam cycle arises from the fact that there is no economically practicable means for recovering the tritium. It exchanges readily with the hydrogen in the steam and would build up in the turbine hall inventory of boiler feed water and steam and ultimately be exhausted to the environment at rates equal to the diffusion and in-leakage rates. Thus these rates must be kept below the total permissible tritium dump to the environment. Although no standards have yet been set for this release, it is anticipated that the maximum allowable release of radioactivity per plant site will be of the order of 1 curie/day. This tolerance translates to 10^{-4} g tritium/day, since tritium is the only source of atmospheric contamination from a fusion power plant under normal operating conditions (except, of course, thermal pollution).

The transport of tritium into the steam cycle can be prevented by interposing a secondary heat transfer loop between the breeding medium and the steam cycle, with the secondary loop located outside the radiation field in the blanket. Since the secondary loop would not be exposed to neutron irradiation, it could use an efficient heat transfer medium like HTS (heat transfer salt), a molten eutectic mixture of sodium and potassium nitrites and nitrates. The breeder fluid would give up energy to the HTS in approriate heat exchangers, and the HTS, in turn, would generate steam in the boilers.

The HTS is a particularly effective barrier for tritium transport because any tritium entering it would be bound up as water, for which the equilibrium pressure of molecular tritium is very low at the temperatures in the steam boilers.

There is a double penalty in capital costs resulting from the introduction of the secondary loop. First, there is the cost of the loop itself. Second, since the intermediate heat exchangers cannot be of infinite size, a finite temperature drop is required to drive them. As a consequence, the steam cycle must operate at a lower temperature and correspondingly lower thermodynamic efficiency, and the overall size of the plant must increase for the same power output.

Another tack that has been taken to minimize the tritium transport to the steam cycle has been to seek coatings for the steam generator tube walls to act as permeation barriers (Bell et al., 1976; Maroni, Van Deventer, et al., 1976). If leaks are as important a mechanism for the tritium transport as permeation through metal walls, this tack might not be helpful. Under normal conditions of operation leaks between the blanket coolant and the steam would necessarily involve flow from the high pressure steam side into the coolant, which would be at much lower pressures. During emergency shutdowns and even planned turnarounds, however, leaks would inevitably lead to contamination of the steam if the coolant contained appreciable tritium pressures.

HELIUM AS COOLANT

As a consequence of all the possible difficulties involved in the use of breeder fluids generally and molten fluorides in particular as blanket coolants, the PRD employs helium gas at about 50 atm pressure for cooling not only the blanket, but all other parts of the reactor system. Some tritium diffuses into the helium from the blanket and from surfaces in the divertor system, but its further diffusion into the steam cycle or elsewhere is prevented by maintaining a modest oxygen pressure in the helium such that any hydrogenic species will oxidize to water on entering the helium. The effective tritium pressure is easily maintained below 10^{-16} atm by holding the water pressure in the helium at 10^{-5} atm through the use of molecular sieve adsorbents or other gas-drying agents operating on a drag stream from the helium loop (Johnson 1975). Recent studies at Oak Ridge (Watson, Bell et al, 1976) have shown that the oxidation of the permeating hydrogen will have to be promoted catalytically.

Helium has the advantage of being indifferent to the neutron flux in the blanket, and it is non-corrosive to conventional metals and alloys. The device of using modest oxygen pressures to control tritium pressures however, does not work with refractory metals. Indeed, even the trace amounts of oxygen in commercially pure helium result in severe corrosive attack on metals like niobium (DeVan, 1972). For these systems, the tritium control in the helium would have to be effected by maintaining a low hydrogen pressure in the helium under swamping conditions, that is, high ratio of hydrogen (or deuterium) to tritium. Unfortunately, this tactic is constrained by the tendency for hydrogenic species to embrittle metals, and for refractory metals the tolerable hydrogen pressure is relatively low.

Helium also has another serious disadvantage as coolant because of its low heat capacity. In order to carry off energy at high rate, it must be pumped at very high pressures and very high velocities. For the PRD, the pumping energy required for the coolant helium exceeds 4% of the gross output of the machine, 11.9% of the net electric power output.

TRITIUM RECOVERY

First generation fusion power plants based on the D-T fuel cycle must breed their own tritium to be economic. Estimated costs of tritium produced by current methods of manufacture range upwards from $3 000/g, which is too expensive for competitive power generation.

Since the transmutation of lithium by neutron bombardment is the only practicable means for producing tritium, the basic fuels for D-T fusion reactors are deuterium and lithium. Regardless of the particular form of the breeder, whether liquid metal, salt or solid, the breeding blanket must be of the order of 1 m thick to ensure net breeding. Since the minor diameter of the reactor must be large enough for adequate plasma residence time, the total volume of the breeder blanket must be large and the associated mass of breeder therefore is also large. Thus, the problem of tritium recovery is one of extracting amall amounts of tritium from very large amounts of breeding medium under conditions that admit of quite complete collection and purification of the tritium. For example, in the PRD 0.585 kg of tritium is produced daily in 1.86×10^6 kg salt in the blanket.

The recovery must be effected at low concentrations to minimize ultimate permeation losses of tritium to the steam cycle and to avoid excessive solution of tritium in structures and subsequent embrittlement of the structures. Another constraint, of course, is the cost, which is a concommittant of availability. Every kilogram of tritium tied up in inventory in first generation machines, that is, before cheap tritium is available from fusion machines, represents a capital commitment of, say, $3 million.

At very large tritium inventories, there is not only a correspondingly increased hazard but also the natural radioactive decay of tritium to helium-3 will adversely affect the doubling time for tritium breeding. Tritium has a half-life of 12.3 yr, which corresponds to an instantaneous loss rate of 5.62%/yr. Thus, a plant with a total tritium inventory of 100 kg would lose 15 g/day through decay. Most conceptual designs for fusion power plants fix the tritium inventory at lower values with the fuel reserve stockpile as the major element.

A more serious aspect of the radioactive decay of tritium is the deleterious effect of the helium formation within metal structures in exacerbating the embrittling characteristics of the tritium.

For all of the foregoing reasons, the tritium concentrations in the breeding media are held to very low values, thereby complicating the recovery process.

TRITIUM RECOVERY FROM LITHIUM METAL

Tritium in liquid lithium readily forms lithium tritide, which, though relatively insoluble by normal standards, is much too soluble for fusion systems to permit tritium recovery by precipitation in a cold trap, even at temperatures just above the melting point of the lithium (179°C).

Although some recent evidence suggests that trace amounts of the tritide may be found in the vapor phase, for all practical purposes the vapor-liquid equilbrium characteristics may be defined in terms of the dissociation reaction

$$LiT = Li + \tfrac{1}{2} T_2 \qquad (14)$$

The resulting expression for the solubility of tritium in lithium is of the form

$$C = K_s(p)^{1/2} \qquad (15)$$

K_s is called the Sievert's law constant (Maroni, 1974). Experimental studies at the Argonne National Laboratory (Maroni, Cataway, et al., 1976) permit the evaluation of K_s for all three hydrogen isotopes up to temperatures as high as 900°C. Unfortunately, the vapor pressures of tritium at low concentrations are so low relative to the vapor pressure of the lithium at any practicable temperature that distillation cannot affect a separation except in enormous equipment. Indeed, it can be inferred that the lithium-tritium system will form an azeotrope at tritium concentrations of about 10^{-4} atom fraction, which is a higher concentration than is tolerable.

Fraas (1973) for his high temperature potassium topping cycle fusion plant has proposed that tritium be recovered from the lithium by taking advantage of the high permeability of niobium to tritium at high temperatures. In a critical study of the problems of handling tritium in fusion reactors, Watson (1972) concluded that permeation might be used for systems involving conventional metals like stainless steel, but that it could not be used for refractory metal systems because the downstream tritium pressure could not be maintained low enough to drive the permeation process. With stainless steel, on the other hand, a small concentration of oxygen on the pressure side of the permeation window would tie up the tritium as water and maintain vanishingly small tritium pressures. Recent experimental results on the permeation of hydrogen through stainless steel reported by Axtmann et al. 1976), however, suggest that even with zero pressure tritium on the downstream side, the permeation rates at low driving pressures would be too low to be practicable.

The problem here is that the permeation appears to be linearly dependent on the driving pressure rather than dependent on the square root of the pressure as occurs at higher driving pressures according to Richardson's expression

$$J = \frac{P}{d}\,(P_1^{1/2} - P_2^{1/2}) \qquad (16)$$

where P is the permeability constant, and the p are the upstream and downstream tritium pressures. In the Axtmann study, the linear law was found to hold for driving pressures as high as 10^{-2} torr. Since these results were obtained for hydrogen permeation from pure gas phase to gas phase of pure hydrogen, there are no mass transfer effects in the gas phases, and the observed rates may be presumed to be dependent solely on surface phenomena and transport within the metal. It might be argued that the fluxing action of fluids like liquid lithium and molten fluorides on typical metal surfaces would mitigate surface resistances to hydrogen permeation. However, Briggs (1971) in an analysis of the observed distribution of tritium throughout the equipment involved in the molten salt reactor experiment at the Oak Ridge National Laboratory found that the amounts were one thousand times lower than would be predicted from Richardson's equation.

It seems reasonable to conclude that permeation may be at best a marginal means for recovering tritium from liquid lithium unless fairly high tritium concentrations are permissible in the lithium. Unfortunately, the refractory metals, which are best suited to permeation methods, are the least tolerant of tritium from the standpoint of metal embrittlement. Clearly, we need a firm experimental base for appraising the possibilities here and also for gauging reliably the permeation losses throughout the reactor system. A detailed review of the permeation problem as it relates to fusion power development is given by Stickney (1972).

In principle, tritium should be recoverable from lithium by chemical absorption using metal getters that have a higher affinity for tritium than does lithium and that are insoluble in lithium. A number of metals including yttrium and zirconium are possibilities. Hubberstey et al. (1976) have undertaken preliminary experiments using yttrium sponge for extracting protium and deuterium from liquid lithium. A potentially serious difficulty with this mode of tritium recovery is the structural stability of the getter on repeated thermal cycling resulting from periodic regeneration of the getter.

An alternative process for tritium recovery is liquid-liquid solvent extraction. Two different schemes have been proposed for this approach, one using molten salts as the solvent, and the other using molten metal alloys.

Based on studies at the Argonne National Laboratory, Maroni, Wolson, and Staahl (1975) propose the use of eutectic mixtures of lithium chloride and potassium chloride as the solvent. The lithium tritide in the lithium is preferentially dissolved in the salt with a volumetric distribution coefficient of about 2. Their preliminary appraisal indicates that the process should be economically feasible provided their proposed but untested scheme for electrolytic recovery of the tritium from the solvent is practicable. Another potential difficulty aside from the ultimate recovery of the tritium from the salt is the possibility of contamination of the lithium by the solvent salt.

Workers at the Los Alamos Scientific Laboratory propose the use of low melting point eutectic mixtures of metals containing yttrium, lanthanum or cerium (Anderson et al., 1976). This process too is in a very preliminary state of investigation, but the distribution coefficient for tritium between these solvents and lithium exceeds 100. Potential difficulties lie in the recovery of the tritium from the solvent and possible contamination of the lithium by elements of the solvent. The process is not strictly a solvent extraction, since the basis for the separation is a chemical competition for the formation of tritides in the two phases.

Typical targets for maximum tritium concentrations in liquid lithium are 0.1 p.p.m. for systems operating at 1 000°C and 1 p.p.m. for systems operating at 500°C, the former depending on permissible permeation rates through refractory metals and the latter dictated by tritium inventory considerations. It is not clear that any of the various schemes for tritium recovery now under consideration can achieve the desired tritium concentrations practicably. Given the many advantages of using liquid lithium as breeder, there are obviously strong incentives to find a suitable separation process. What is needed is chemical engineering ingenuity of the highest order.

TRITIUM RECOVERY FROM MOLTEN SALTS

Although the recovery of tritium from molten fluorides appears to be much simpler than for other breeders like lithium metal, the problem is not an easy one.

As pointed out previously, the tritium born of the fusion reactions in the blanket will exist as tritium fluoride for the most part, with some present as molecular tritium and free tritium at the metal walls. For a power system like the PRD, formation of molecular tritium as a result of interaction between metal wall and tritium fluoride could be regarded as an advantageous feature of the tritium recovery system inasmuch as this tritium would permeate the walls and be recovered economically and completely from the coolant helium system. As a matter of fact, if all of the tritium production in the blanket were to follow this path, and if the attack on the metal walls were uniform over the whole metal-salt interface, the walls would not require replacement before 30 yr of operation. Since it would be unlikely that the corrosive attack could be maintained uni-

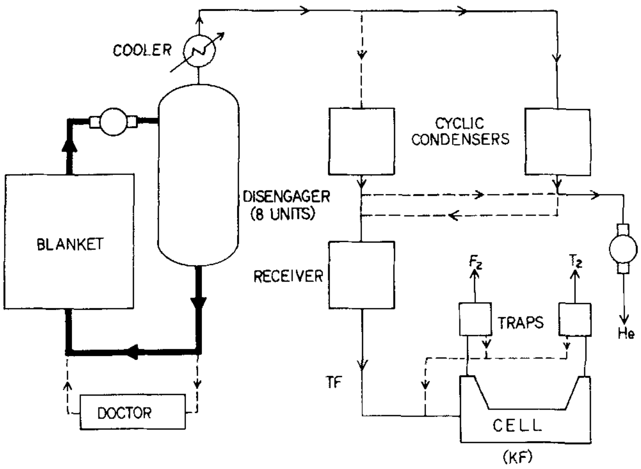

Fig. 6. Salt processing system.

form, it would be desirable to minimize the corrosion by proper choice of metal walls and by doctoring up the salt with appropriate buffering agents.

DeVan (1976) has operated a flibe loop in 316 stainless steel at 700°C for more than 4 500 hr with no detectable corrosion by the simple expedient of inserting a beryllium finger in the loop. Normally at these temperatures the chromium in the metal dissolves in the salt. On thermodynamic grounds it would be reasonable to expect that the beryllium would inhibit the formation of TF and hence any corrosive attack on the walls of a breeder salt system. The presence of the beryllium, of course, would be advantageous from the standpoint of the system neutronics by virtue of its neutron multiplication capability [Equation (12)].

Actually, if the molten salt is flowing through the blanket in laminar flow as is proposed for the Princeton machine, the maximum fraction of the tritium production which could permeate the metal walls under steady state conditions is less than one third. Maienschein (1974) has shown that this maximum obtains regardless of the nature of the metal wall and the processes occurring at the salt-metal interface because the rate limiting step is the diffusion of the tritium and tritium fluoride through the salt. Thus the bulk of the tritium recovery problem is the separation of the tritium or tritium fluoride from the molten salt under conditions which concentrate the tritium or its fluoride. This separation is accomplished in the PRD by the system shown in Figure 6. Table 5 lists the important characteristics.

Hot breeder salt from the reactor blanket recirculates continuously at a rate of 4 × 10^6 kg/hr through eight desorbers or disengagers distributed in parallel around the reactor system. In the PRD design the salt is sprayed in the form of small droplets into the desorbers which are maintained at pressures below 10 N/m^2. Tritium fluoride and helium gas together with small amounts of vaporized salt go off overhead, and the bulk of the salt collects at the bottom of the desorber and is pumped back through the blanket. A small drag stream on the salt is drawn off intermittently and run through a salt doctor to remove impurities and add buffering and other agents to maintain the desired salt composition.

Maienschein (1974) and Talbot (1976) have shown that spray columns are likely to be marginal desorbers for recovering tritium from the salt. Getz (1976) has shown that packed column desorbers with the salt percolating down through conventional packing should suffice, even for individual disengagers no larger than 1 m in diameter and 3 m high.

If the blanket is operated to produce molecular tritium rather than tritium fluoride, it would be necessary to introduce a small flow of fluorine into the desorber offgas to burn the tritium to the fluoride. This combustion should proceed rapidly at the temperatures exit the desorbers and not require elaborations of equipment or catalyst.

TABLE 5. SOME CHARACTERISTICS OF TRITIUM RECOVERY SYSTEMS ON BREEDER SALT LOOP

Salt flow through disengagers	4.0 × 10^6 kg/hr
Concentration of TF in salt, mole fraction	
Entering disengagers 2.1 × 10^{-7}	
Exit disengagers 1.4 × 10^{-7}	
Composition of gas exit disengagers, N/m^2	
TF	10.1
He	15.6
Salt	13.2
Tritium recovery	0.548 kg/day
Helium ash discharge	1.120 kg/day

TRITIUM REGENERATION

As can be seen from Figure 6, the small flows of gases leaving the top of each disengager are drawn first through a small cooler to knock out salt vapor and entrained salt, then through a small trap cooled with boiling nitrogen to crystallize out all the tritium fluoride.

The nitrogen cooled traps are thawed out periodically, and the molten tritium fluoride is dropped to small electrolytic cells wherein melts of TF and KF are electrolyzed to T_2 and F_2 (Johnson, 1972). The respective gases are passed through cold traps to remove any TF, the T_2 then flowing to the primary fuel loop processing system and the F_2 flowing to storage.

THERMAL BALLAST

Since tokamak reactor systems must operate cyclically, it is necessary to provide some kind of energy ballast to ensure that the delivery of electrical energy to the power grid is at constant rate. In the PRD, the reactor is on stream about 100 min and off 3 min. By maintaining an inventory of breeder salt three times the nominal holdup in the blanket, the salt can serve as a thermal ballast for the plant. During the off stream part of the cycle, when no energy is being produced, the salt flow bypasses the desorbers and merely circulates the blanket and storage system, giving up its energy to the coolant helium. The average decline in salt temperature is only 50°C for an inventory of 6 × 10^6 kg salt.

TRITIUM RECOVERY FROM SOLID BREEDERS

The principal problems with the use of solids as tritium breeders have to do with cooling and tritium recovery. There seems to be no problem in achieving acceptable breeding ratios per se except that the requirement of high lithium density conflicts with the need for flow channels for the coolant and short diffusion paths for the tritium.

Helium is the typical coolant; hence high gas pressures must be accommodated in the blanket. Since tritium diffusion rates in solids are much lower than in liquids, the solids must be sufficiently comminuted to ensure transport of the tritium into the helium.

Wiswall and Wirsing (1976) have shown experimentally that tritium can be removed readily from solid LiAl, $LiAlO_2$, and Li_2SiO_3 in the form of powder or small granules. For particles not larger than 50 μm maintained at temperatures not less than 800°C, they estimate the tritium inventory in a reactor blanket could be held to 10 curies/MW(e), that is, 10^{-3} g tritium/MW of net electricity generating capacity. Tanaka et al., (1976) have conducted similar experiments on pebbles of lithium oxide. This compound has the happy property that its lithium density is higher than that of pure liquid lithium.

Guggi et al., (1976) present data on tritium release rates from $LiAlO_2$. Owen and Randall (1976) describe measurements at the Savannah River Laboratory for the same system and also for the distribution of helium and tritium throughout irradiated lithium-aluminum alloy. They present equilibrium data over wide temperature ranges for all three hydrogen isotopes in contact with lithium hydrides plus aluminum, lithium aluminum alloy, and irradiated lithium aluminum alloy. Veleckis and Maroni (1976) summarize results obtained at the Argonne National Laboratory on the thermodynamic properties of solutions of hydrogen isotopes in lithium and in lithium aluminum alloy. They show that the dissociation pressure of hydrogen in solution in the beta phase of 50-50 molar lithium aluminum alloy is 25 fold greater than that for hydrogen in lithium at the same atom fraction hydrogen and same temperature (600°C).

The original incentive for considering aluminum containing systems was to develop reactor systems with low enough residual radioactivity after normal operation to permit direct, as opposed to remote, maintenance. The principal penalty deriving from reliance on aluminum in reactor structures is that the operating temperatures are severely limited, and overall thermodynamic efficiencies cannot be high. Although some clever designs have been proposed to circumvent the temperature limitations (Powell et al., 1973), there remain formidable engineering problems to meet reliably the demanding constraints imposed on practicable power generators.

TRITIUM RECOVERY FROM HELIUM

Helium has been proposed as the blanket coolant for a wide variety of hypothetical designs. It is probably mandatory for blankets employing solid breeders or molten salt breeders, and even for blankets using other coolants helium is usually proposed as the purge gas in piping jackets, valve shrouds, and the like devices needed to prevent escape of tritium to the immediate environment. Thus, essentially, all reactor design systems will confront the problem of recovering tritium from helium, typically at very low tritium concentrations and under conditions permitting nearly total recovery and easy regeneration as pure tritium.

For structural materials like stainless steels which can tolerate oxygen in the helium, the recovery system proposed for the PRD affords the simplest solution. Johnson (1975) has described the process in some detail. A small oxygen pressure corresponding to perhaps 50 p.p.m. is maintained in the coolant helium with the result that any tritium or other hydrogen isotope entering the helium at the temperatures prevailing in blanket or divertors is oxidized to water. A drag stream comprising less than 1% of the total helium flow (5×10^3 kg/hr in the PRD) is taken off the coolant loop at the point of lowest helium temperature in the loop (66°C) and circulated through mole sieve adsorbers without further cooling and at the operating pressures of the coolant, 50 atm (5 MPa). The relatively small adsorbers operating cyclically for regeneration have no difficulty in maintaining the water pressure in the helium below 10^{-5} atm (1 Pa). In the regeneration step, a given adsorber is heated to 260°C and pumped down to 2×10^{-6} atm (0.2 Pa). The gases leaving the adsorber are drawn through small nitrogen cooled traps to condense out the water. Some of the water will contain protium as a result of back permeation from the steam side of the boilers into the helium. There will also be some deuterated water from permeation in the divertors. These isotopes are routinely separated in the fuel processing system on the primary fuel loop after the trapped out water is electrolized to hydrogen isotopes and oxygen.

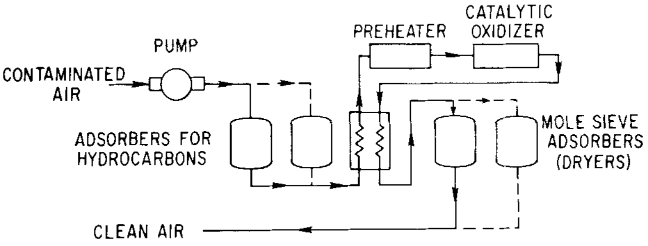

Fig. 7. Tritium control system for secondary containment.

While this process for recovering tritium from helium cannot be studied fully in the absence of intense neutron fluxes, it would be reasonable to infer that the effect of the neutrons would largely ensure the establishment of thermodynamic equilibrium within the helium and hence ensure that all hydrogen would be oxidized to water. The rest of the problem is the drying of dry gases in a manner that permits complete recovery of the water.

For refractory metal reactor systems, the helium would have to be maintained under reducing rather than oxidizing conditions, and getters capable of removing hydrogen isotopes in the parts per billion range would be necessary. Schoenfelder and West (1976) have evaluated a number of getters including pure metals, alloys, supported metals, and a high molecular weight ether.

Gettering the hydrogen isotopes is only half the problem; the other half is regenerating the isotopes. This step is apt to be slow (Watson, 1972), but the isotopes come off as DT, TH, etc., and as such can proceed to the primary fuel loop processing system for purification.

TRITIUM CONTROL

The problem of tritium control, that is, the restriction of releases to the environment, has already been alluded to, albeit briefly. In the PRD, the problem is dealt with in a number of ways.

First, the turbine hall is isolated from the reactor building so that the only tritium entering the turbine hall is the closely controlled amount that leaks or permeates into the steam system from the coolant helium. Second, all lines or other elements carrying streams containing tritium are enclosed in jackets with a low pressure helium purge to interpose a tritium collecting barrier between the operating units in the reactor building and its atmosphere. A similar purge is maintained between the blanket and the radiation shield protecting the magnets. Fourth, the atmosphere in the reactor building is monitored continuously for tritium, and in the event of a tritium release into that atmosphere, the emergency system shown schematically in Figure 7 is activated. The reactor building is essentially a hermetically sealed unit. Fifth and finally, the processing system on the primary fuel loop, including the fuel stockpile, and hence most of the tritium inventory in the entire plant, is enclosed in a small bombproof, quakeproof cell.

The technology of tritium control is pretty well in hand, and no surprises are anticipated. Useful perspectives on this technology are given by Hickman (1975) in a discussion of tritium problems in fusion reactor systems and by Garber (1976) in a detailed description of the tritium handling concepts used for the TFTR.

FISSION-FUSION HYBRIDS

In the early days of the fusion program, proponents of fusion power would cite cleanliness, inherent safety, and inexhaustable fuel supply as unique advantages of fusion power. These supposed advantages contrasted sharply with the corresponding properties of fission power machines with their escalating production of long-lived radioactive wastes, their potential for disastrous runaway, and their reliance on scarce uranium reserves. As a consequence, there was a marked reluctance on the part of many in the fusion community to contemplate even remotely the possibility of coupling fusion and fission machines in effective synergism. More recently, our growing familiarity with the problems that beset fusion breeds a respect for them, and the idea of fission-fusion hybrids now commands a growing interest and activity.

Fusion reactions can produce a plethora of fast neutrons which can be used to drive fission reactions. Since the energy release per fission is tenfold that from a fusion event, the net result of hydridization, in principle at least, is a substantial multiplication of energy production overall for a given size of fusion reactor. Furthermore, as Lidsky (1975) points out hybrid machines can be used to breed fissile fuels, namely, plutonium from ordinary or depleted uranium or uranium-233 from thorium, and they can be used to transmute objectionable long-lived radioactive wastes to relatively harmless materials.

As a consequence of these potential advantages, a number of laboratories have undertaken studies of hypothetical fission-fusion hybrid machines. For example, the Princeton Plasma Physics Laboratory is designing a machine based on the TFTR but including a blanket in which tritium is bred in a manner similar to the PRD, and plutonium is produced from uranium and subsequently burned.

The original concept was a machine not much larger than the TFTR, and like the TFTR, driven by energetic neutral beams of deuterons; that is, the injected energy would be necessary to sustain the fusion reactions. A blanket surrounding the reactor would simultaneously breed tritium from lithium and plutonium from uranium and derive further energy from fission of the plutonium. Although the study has not been completed, the results so far (Tenney, 1976) suggest that the machine will have to be much larger than the TFTR, more nearly approaching the dimensions of the PRD, and it will probably be best suited to power generation rather than plutonium production.

The chemical engineering problems posed by the hybrid differing from those of the fusion machine alone include the reprocessing of partially converted uranium, presumably from solid slugs, the recovery of tritium from systems now complicated by the presence of uranium and plutonium, and the removal of radioactive fission wastes. These problems, though by no means trivial, are of secondary importance in exploring optimal combinations of fusion and fission capabilities. The problems of the plasma physics still dominate the constraints on possible designs.

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NOTATION

C = concentration of tritium in the liquid
d = thickness of the metal membrane
J = permeation rate
k = Boltzmann constant
K_S = Sievert's law constant
M = divalent metal
n_i = number density of ion i, cm^{-3}
p = total pressure
R = number of fusion events/cm^3 s
T = absolute temperature
v = particle velocity, cm/s
σ = cross section for the reaction, the reaction probability per target nucleus per unit incident flux, cm^2
$\overline{\sigma v}$ = product of the cross section for the reaction averaged over the velocity distribution

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