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A LIMITER/VACUUM SYSTEM FOR PLASMA IMPURITY CONTROL AND EXHAUST IN TOKAMAKS*

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Summary

A detailed design of a limiter/vacuum system for plasma impurity control and exhaust has been developed for the STARFIRE tokamak power plant. It is shown that the limiter/vacuum concept is a very attractive option for power reactors. It is relatively simple and inexpensive and deserves serious experimental verification.

Introduction

Previous reactor design studies have shown that plasma impurity control and exhaust is one of the most difficult systems in tokamaks. Therefore, the STARFIRE study[1] has devoted a significant effort to the development of a credible and attractive design for this system. This paper develops the design concept and presents a summary of the performance characteristics of the reference impurity control and exhaust system in STARFIRE. More details are given in Ref. 1.

An assessment of the impurity control and exhaust system based mostly on previous work in this area identified five basic problems: (1) high heat load on the particle collection medium; (2) high tritium inventory in the fueling system and vacuum pumps; (3) very large vacuum-pumping speed requirements; (4) significant neutron and gamma-ray streaming through the vacuum ducts leading to high heat loads on the pumping cryopanels and difficult shielding requirements; and (5) engineering complexity inherent to some specific concepts for plasma ash removal. The STARFIRE approach to solving these problems is discussed below.

The origin and solution to the first problem of high heat load on the particle collection medium are highly dependent on the characteristics of plasma operation. In steady-state, the alpha power plus any auxiliary heating power must be removed from the plasma region. In STARFIRE, the alpha power is 700 MW and the rf power is 90 MW, giving a total of 790 MW. In conventional designs, only less than half of this energy is radiated leaving more than 400 MW to be transported to the particle collection medium. Previous designs for divertors showed that the surface area of the particle collection medium is limited to $\sim 20\ m^2$. For these designs, the average heat load would be $> 20\ MW/m^2$ and, given the fact that the particle heat load drops exponentially across the scrape-off region, the peak heat load would be $> 50\ MW/m^2$. Such an extremely high heat load is beyond the capability of any suitable structural material. The STARFIRE approach to solving this problem consists of two parts:

(a) Enhancing plasma radiation to reduce the transport power to the particle collection medium. This is accomplished by injecting small amounts of high-Z material (iodine) along with the DT fuel. Most of the alpha energy is thus radiated to the first wall which has a large surface area.

(b) Increasing the surface area of the collection medium. One convenient method of accomplishing this is to minimize the angle between the direction of incidence of the charged particles and the surface of the collection medium. There are limitations on the size and position of the collection medium, which vary from one impurity control concept to another.

The second, third, and fourth problems of high tritium inventory, large pumping speeds, and troublesome radiation streaming are strongly interrelated as to the origin of the problems and the approach to solving them. Previous studies strived to achieve a high helium removal efficiency approaching unity. This removal efficiency is defined as the probability that a particle diffusing out of the plasma will be pumped rather than reflected into the plasma. By requiring a helium removal efficiency of ~ 1, the fraction of deuterium (D) and tritium (T) recycled into the plasma (reflection coefficient) becomes low and the gas load to the vacuum pumps increases. A low tritium reflection coefficient results in a low tritium fractional burnup and an increase in

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the tritium inventory requirements in the fueling system. The increase in the DT gas load in the vacuum pumping system leads to an increase in (a) the tritium inventory in the vacuum pumps; (b) the required pumping speed; and (c) the required capacity of the vacuum pumps. The intensity of radiation streaming is critically dependent on the size and shape of the vacuum ducts. Again, requiring high helium removal efficiency requires a high conductance vacuum pumping system, which can only be realized by large-size vacuum ducts with no significant bends.

A key part of the STARFIRE solution to these problems is to design for only a modest helium removal efficiency. As discussed in Ref. 1, it can be shown that steady-state plasma operation is achievable with a helium removal efficiency as low as 10-20%. The penalty of such a low removal efficiency is a high alpha particle equilibrium concentration in the plasma. In STARFIRE, this is compensated for by a modest increase in the strength of the toroidal field to keep the fusion power the same. It should be noted that the significant charge-exchange with hydrogen tends, in general, to make the tritium removal efficiency lower than that for helium.

The solutions outlined for the four problems above can be applied to any design concept for impurity control and exhaust. However, the degree of success varies considerably from one concept to another. This degree of success is an important figure of merit in selecting a design concept.

The fifth problem of engineering complexity is specific to the particular design concept selected for the impurity control and exhaust system. The magnitude of the problem is greatly dependent on the configurational and component requirements of the specific concept and how they integrate with the rest of the reactor system. Divertor and divertorless concepts were surveyed. Present design concepts for poloidal and bundle divertors are found to be inherently complex. Specifically, they require magnets, enhance radiation streaming, complicate maintenance, and significantly increase the physical size of the reactor. Therefore, it seems prudent to seriously explore divertorless concepts.

An evaluation of divertorless schemes shows that the "pumped limiter" (also called limiter/vacuum system) is an attractive concept with many inherently simple features that are very desirable in a commercial power reactor. Among the advantages of the limiter/vacuum system, as compared to divertors, are:

(1) It is a mechanical system that does not require magnets.

(2) It has minimal requirements on space; the limiter fits naturally into the scrape-off region.

(3) Because of its location inside the first wall, the surface area available for the limiter is relatively large, thus permitting operation at reasonable heat fluxes.

(4) The system is flexible enough to permit designing for low hydrogen removal efficiency; this leads to higher tritium fractional burnup, low tritium inventory, reduced gas loads, and more attractive requirements for the effective pumping speed.

(5) The limiter/vacuum system can be designed to dramatically reduce radiation streaming.

(6) The limiter can be replaced simultaneously with the first wall with no special maintenance requirements.

(7) The system is simple and inexpensive. This feature is not only attractive for reactor maintainability and economics, but it also means that the physics and engineering testing necessary to qualify the concept can be done in present facilities, in a relatively short time and at a modest cost.

Several variants of the limiter were discussed earlier in the literature (see for example, Ref. 2-7). The present work represents the first comprehensive attempt to develop a detailed design supported by physics and engineering analyses.

2. Reference Design Summary

A serious effort has been made in the STARFIRE study to develop a plasma impurity control and exhaust system that satisfies the following goals: (1) have manageable heat loads in the medium where the alpha and impurity particles are collected; (2) have a reasonable and reliable vacuum system that minimizes the number and size of vacuum ducts; (3) have a high tritium burnup to minimize the tritium inventory in the fuel cycle; and (4) have engineering simplicity compatible with ease of assembly/disassembly and maintenance.

These goals are found to be best satisfied by a toroidal limiter/vacuum system together with a beryllium coating on the first wall, limiter, and all other surfaces exposed to the plasma. In order to minimize the heat load to the limiter, most of the alpha-heating power to the plasma is radiated to the first wall, by injecting a small amount of high-Z material, e.g. iodine, along with the DT fuel stream. The iodine atoms enhance the line-and-recombination radiation over most of the plasma volume. The helium removal efficiency of the limiter/vacuum system is intentionally kept low for three reasons: (1) to reduce the heat load on the limiter; (2) to simplify the vacuum system and reduce radiation streaming; and (3) to minimize the tritium inventory tied up in the vacuum and tritium processing systems. The major features of the STARFIRE impurity control and exhaust system are summarized in Table 1.

Table 1. Major Features of STARFIRE Impurity Control/Exhaust

- A limiter/vacuum system
  - One toroidal belt-type limiter centered around midplane
  - Simple, inexpensive, credible engineering
- Low-Z coating (beryllium) on all surfaces exposed to plasma
- Enhanced plasma radiation
  - To reduce heat load at collection plate
  - Achieved by injecting small amount of iodine
- A low helium removal efficiency (25%)
  - Much simpler vacuum system
  - Less radiation streaming
  - High tritium burnup, low tritium inventory
  - Penalty: Modest increase in toroidal field (0.85 T on axis)
- Simple vacuum system
  - Limiter duct penetrates blanket leading to a plenum region between blanket and bulk shield
  - Significantly reduced radiation streaming; less shielding and lower nuclear heat load in cryopanels

Figure 1 shows a cross section through the limiter, the limiter slot, the limiter duct, and the plenum region. The limiter consists of 96 segments that form one toroidal ring centered at the midplane and positioned at the outer side of the plasma chamber. This location was selected because: (1) it is the least likely place for a thermal energy dump from a plasma disruption; and (2) it helps the symmetry in particle and heat loads on the upper and lower branches of the limiter. Each of the limiter segments is 1 m high and ∿ 0.6 m wide. The physical dimensions of the system are shown in Fig. 1. The limiter slot, which is the region between the limiter and first wall, leads to a 0.4-m high limiter duct that penetrates the 0.7-m thick blanket. The limiter duct opens into a plenum region that is located between the blanket and shield and extends all the way around the torus. This plenum region is large enough so that it spreads the radiation leakage from the limiter duct into a larger surface area of the bulk shield. The conductance of the plenum region is large enough to permit locating the vacuum ducts in the bulk shield sufficiently removed from the midplane so that radiation streaming from the limiter duct in the blanket to the vacuum pumps is acceptable. There are 12 vacuum ducts at the top and another 12 at the bottom of the reactor. Each of these vacuum ducts has an equivalent diameter of 1 m and penetrates the bulk shield leading to the vacuum pumps.

The basic principles of how the limiter works are rather simple. Ions that hit the front face of the limiter will be neutralized and reflected back into the plasma. Ions that fall into the limiter slot hit the back surface and are neutralized. Some of the scattered neutrals will directly reach the limiter duct and follow a multiple-scattering path into the plenum region and out the vacuum ducts where they are removed by the vacuum pumps. Other particles neutralized at the back surface of the limiter will scatter back in the direction of the plasma. These neutrals have a high probability of being ionized and returned back to the limiter surface. Calculations show that this trapping or "inversion" effect is so large for helium that ∿ 90% of the helium entering the limiter slot will get pumped. This inversion effect greatly simplifies the limiter/vacuum system design in at least two ways:

(1) Location of the Leading Edge: Since the helium inversion probability is very high, the fraction of particles that enters the limiter slot needs to be only slightly greater than the helium removal efficiency. This permits locating the two leading edges at the top and bottom of the limiter sufficiently away from the plasma edge and inward into the scrape-off region so that the peak heat

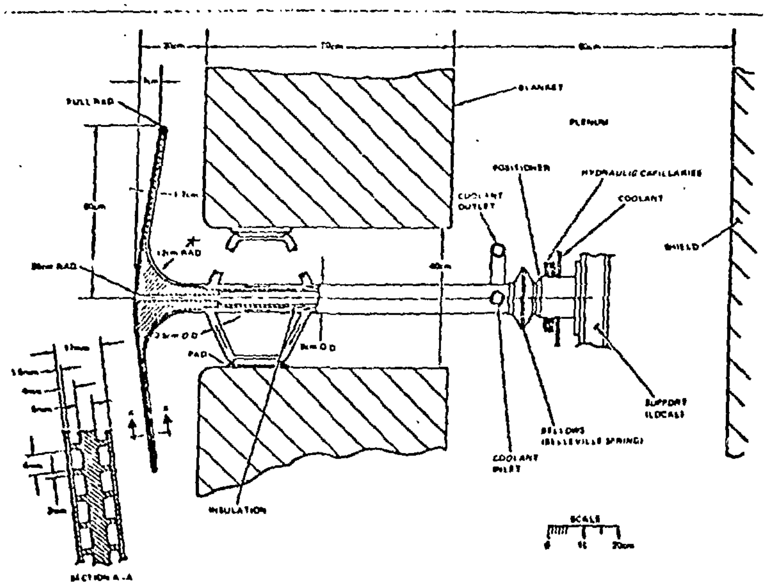

Fig. 1. Cross section of the STARFIRE limiter design.

flux at the leading edge is reasonably low.

(2) Neutral Pressure: This inversion effect causes the neutral gas pressure at the limiter duct to be considerably higher than the neutral pressure around the plasma. Such high pressure greatly simplified the vacuum system design.

Hydrogen can charge-exchange as well as be ionized. These charge-exchange events significantly reduce the inversion probability for hydrogen because the resulting neutral will tend to make its way out of the slot region into the plasma. Therefore, the beneficial effect of higher helium pumping probability and enhanced hydrogen recycling into the plasma is obtainable with the limiter/vacuum system.

The plasma parameters related to the impurity control and exhaust system are shown in Table 2. The design parameters for the limiter are shown in Table 3.

The charged particle flux in the scrape-off region falls off exponentially as e^{-x/δ_p} with δ_p = 10 cm and x as the distance into the scrape-off region. About 28% of the helium particles diffusing out of the plasma will flow into the limiter slot, i.e. between x = 8.7 cm and x = 20 cm. The transmission (pumping) probability for these particles is 0.9 giving an overall helium reflection coefficient R_e = 0.75 and helium removal efficiency $(1 - R_e)$ of 0.25. The reflection coefficient for deuterium/tritium is 0.9.

The transport heat flux on the limiter (due to charged particles) varies as $16\ e^{-x/\delta_E} \sin\theta$ in units of MW/m^2 with x in cm, δ_E = 5 cm, and θ being the angle between the field lines in the poloidal plane (nearly vertical in Fig. 1) and the surface of the limiter. The limiter surface from the tip (at the plasma edge) to the top (or bottom) leading edge is slanted to spread the heat load. The leading edge (region where θ = 0 occurs) forms approximately one-half a cylindrical shell with a diameter of 1.7 cm and extends from x ∿ 7 cm to x ∿ 8.7 cm. The location of the leading edge was determined from tradeoffs between the helium removal efficiency (and the associated toroidal-field margin) and the peak heat flux. The peak transport heat flux is ∿ 3.4 MW/m^2 and occurs at x ∿ 7.8 cm. The average transport heat flux on the region of the leading edge is ∿ 2.2 MW/m^2. The front surface of the limiter (from the tip to the plasma radiation and charge-exchange neutrals. The magnitude of the volumetric nuclear heating depends on the specific limiter material and is in the range of 30 to 80 MW/m^3 for the materials to be discussed shortly.

Water is selected as the limiter coolant because of its good heat transfer characteristics. This choice is consistent with the use of water cooling in the first wall and blanket. The limiter segments are connected so that the coolant passes through two segments (2 passes). The water inlet temperature to the first pass is 115°C and, with a temperature rise of 15°C per segment, the

Table 2. Plasma-Related Parameters

Fusion alpha power (P_α), MW	703
Lower-hybrid power to plasma (P_{LH}), MW	90
Transport power to the limiter, MW	90
Helium production rate, (particles per second)	1.24×10^{21}
Alpha particle concentration (n_α/n_{DT})	0.14
Beryllium (low-Z coating) concentration (n_{Be}/n_{DT})	0.04
Iodine (radiation enhancement) concentration (n_I/n_{DT})	1.0×10^{-3}
Helium reflection coefficient, R_α	0.75
Toroidal-field margin at plasma center, T	0.85
Scrape-off region thickness, m	0.2
Particle confinement time (τ_p), s	1.8
Particle e-folding distance in scrape-off zone (δ_p), cm	10
Energy e-folding distance in scrape-off zone (δ_E), cm	5
Plasma-edge temperature (T_{edge}), keV	1.2

Table 3. Limiter Design Parameters

Coolant	Water			
Reference structural materials	Ta-5W, AMAX-MZC, FS-85, or V-20Ti			
Low-Z coating material	Beryllium			
Total heat removed from limiter, MW (90 MW transport, 56 MW radiation plus neutrals, and 54 MW nuclear)	200			
Average surface heat load, MW/m^2	2.3			
Peak surface heat load,[a] MW/m^2	4			
Coolant inlet temperature, °C	115			
Coolant outlet temperature (2-pass), °C	145			
Coolant pressure, MPa (psia)	4.2 (600)			
Coolant channel size, mm × mm	8 × 4			
Wall thickness, mm	1.5			
Maximum temperature, °C	Ta-5W	AMAX-MZC	FS-85	V-20Ti
Water side	193	182	192	191
Coating side	290	196	404	449

[a]Includes transport load (3.4 MW/m^2) plus load from radiation and charge-exchange neutrals.

outlet temperature is 145°C for the second pass. The coolant pressure is 4.2 MPa (600 psia). The water temperature is kept low to minimize presure stresses. Since the 200 MW of heat removed from the limiter represents only 5% of the reactor thermal power, this heat is used effectively for feedwater heating in the steam cycle without significant loss in thermal efficiency.

A large number of materials were evaluated as to their suitability for the limiter structure. The evaluation included the capability of withstanding high heat fluxes, resistance to radiation damage, fabricability, and compatibility with the surrounding environment. This resulted in identifying four reference alloys as the primary candidate materials. These included a copper alloy AMAX-MZC, and the refractory metal alloys of vanadium (V-20Ti), niobium (FS-85), and tantalum (Ta-5W). Three-dimensional thermal-hydraulic, and stress analyses were carried out for these four materials. A summary of the results is shown in Table 4.

The limiter wall temperature at the coolant side is essentially the same, < 200°C, for all materials with small differences due to axial conduction. At this low temperature, the corrosion rate of these materials in water should be acceptable. The maximum temperature in the structure (coating side) varies from 196°C in copper to 449°C in vanadium reflecting the large difference in the thermophysical properties. The ratio of the effective stress to the yield stress is also shown in Table 4. These results indicate that under normal operating conditions, all of the materials meet the allowable stress criteria of the ASME Code Case 1592. However, only AMAX-MZC and Ta-5W can meet the more restrictive criteria of 0.75 of the yield strength. Since the thermal stress component dominates the total stress in the limiter, the materials with the highest thermal conductivity and lowest thermal expansion will experience the lowest stress. It should be noted, however, that the results in Table 4 are based on conservative assumptions. Furthermore, several modifications in the reference limiter design that can significantly reduce the thermal stress have been identified and are discussed in Ref. 1. Therefore, all the four alloys in Table 4 are considered viable candidates and the selection of one of them must be made based on additional data from future experimental results in areas such as resistance to radiation damage.

The limiter and the first wall are coated with beryllium to eliminate sputtering of the underlying high-Z structural materials. Beryllium is selected as the low-Z coating because its properties make it superior to other candidates. Estimates of the erosion of the beryllium coating were made. The coating on the first wall will erode at a rate of 0.14 mm/yr; therefore, a 1.2-mm coating is adequate for a 7-yr life. The limiter coating will sputter by all ion species with a spatially varying rate. Redeposition of beryllium from the plasma and first wall will also occur. The net effect is that the coating will erode on the wall while it grows on the limiter. The STARFIRE design is developed such that there is no net erosion or growth on the leading edge. This is accomplished by maintaining a beryllium density in the plasma of ∿ 4% of the hydrogen ion density. There will be a net growth of beryllium on the rest of the limiter averaging ∿ 0.6 mm/yr. A simple grinding process in place can be performed if necessary to restore the beryllium coating to its original thickness.

The response of the limiter to off-normal conditions was considered as an integral part of the design. The important off-normal events are: (1) plasma disruptions; and (2) loss-of-coolant flow. The concerns with plasma disruptions are the thermal energy dump and the induced electromagnetic forces. The limiter is intentionally located at the outer side of the plasma and centered around the midplane, where a plasma energy dump is least likely. However, in the unlikely event that a plasma thermal energy dump on the limiter occurs, only the coating will be affected. The rate of ablation of beryllium is small enough that several disruptions per year with the thermal energy dump on the limiter can be tolerated.

The electromagnetic forces will always be induced in the limiter in the case of a plasma disruption regardless of where the plasma energy dump occurs. Three electromagnetic effects are produced, with the magnitude strongly dependent on the plasma disruption (current decay) time. The first is a uniform pressure, acting on the outside panels of the limiter. For a plasma disruption time of > 10 ms, the maximum induced stress due to this uniform pressure is 0.6 MPa (90 psi), which is a small fraction of the yield stress for the copper, tantalum, niobium, and vanadium alloys. The second effect is a force tending to bend the limiter arm about a toroidal axis. Accommodating this force required an iterative process in the limiter design. In particular, providing a thick root for the limiter (see Fig. 1) was found necessary to reduce the moment arm and the magnitude of the force. With the present reference design, the maximum bending stress is ∿ 154 MPa (22,000 psi), which is < 40% of the yield stress for the reference structural materials when the plasma disruption time is > 10 ms. The third electromagnetic effect is a torque that tends to twist the limiter about a radial axis. For a plasma disruption time of 10 ms, the maximum torque is 46 KN-m resulting in an effective stress which is < 60% of the yield stress for all of the four primary structural materials. The magnitude of these forces and torques is reduced substantially at longer, and perhaps more realistic, plasma disruption (current decay) times. The reference limiter design can withstand the electromagnetic effects without any permanent deformation for an unlimited number of plasma disruptions.

Table 4. Thermal/Stress Analysis of Candidate Limiter Materials[a,b]

	Temperature (°C) Outer	Temperature (°C) Inner	Maximum Effective Stress (MPa)	Yield Stress (MPa)	Effective Yield
Tantalum, Ta-5W	290	193	249	342	0.7
Niobium, FS-85	404	192	370	370	1.0
Vanadium, V-20Ti	449	191	537	452	1.2
Copper, AMX-MZC	196	182	178	431	0.4

[a] Coolant: Pressure = 600 psi, T_{in} = 115°C, T_{out} = 145°C
Channels = 4 × 8 mm, 1.5 mm thick at outer side.

[b] Peak heat load = 4 MW/m².

A loss-of-coolant accident were also analyzed and no difficult problems could be anticipated. Figure 2 shows the temperature response of the beryllium coating and the limiter structure (Ta-5W) to a loss-of-coolant event. It was assumed that the response time of the instrumentation and central system is ∿ 2 s and that the fusion power decays linearly during a 10-s shutdown time. The outer surface of the limiter was assumed to radiate thermal energy to the surrounding first wall that is maintained at 400°C. As shown in Fig. 2, a maximum temperature of < 700°C is reached in ∿ 4 s, after which time the temperature starts to go down.

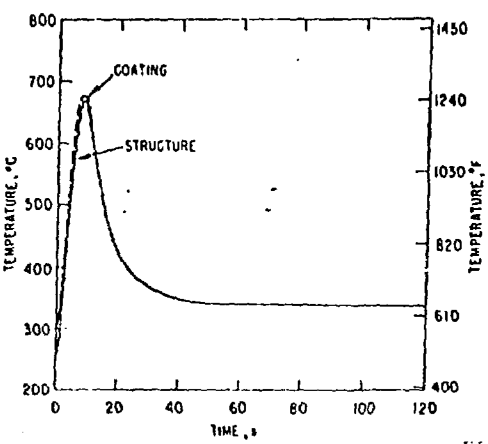

Fig. 2. Temperature response of beryllium coating and limiter structure to loss-of-coolant.

The major components of the vacuum system are shown in the STARFIRE reactor cross section of Fig. 3. The design parameters for the vacuum system are given in Table 5. The vacuum system consists of the limiter slots, limiter ducts, plenum region, vacuum ducts, and vacuum pumps. There are 48 compound cryopumps operating on 24 vacuum ducts. Two pumps are provided on each duct so that regeneration can be accomplished during plasma operation. Each pump has a rated helium pumping speed of 120 m^3-s^{-1}. The vacuum system is designed to produce a base pressure of ∿ 1.3 × 10^{-6} Pa (10^{-8} torr). Tritium inventories in the pumps are minimized by the achievement of a very high tritium fractional burnup (42%) and by minimizing the pump regeneration time (2 hr). The maximum tritium inventory in a single pump is only 2.6 g.

3. Conclusions

The impurity control and exhaust system is one of the key components in a fusion reactor. It has a substantial impact on the engineering simplicity, reliability, maintainability, economics and safety of the power plant. Divertors and divertorless options were surveyed. It was concluded that the limiter/vacuum (also called "pumped" or "active" limiter) concept is a very attractive option for power reactors. It is relatively simple and inexpensive and deserves serious experimental verification.

The main advantages of the limiter/vacuum system, as identified in STARFIRE, are: (1) it is a mechanical system (it doe not require magnets); (2) it has minimal impact on access and breeding blanket space; (3) it can be designed to dramatically reduce radiation streaming; (4) the surface area available for particle collection is relatively large; and (5) it permits designing for higher tritium fractional burnup and lower

Table 5. Vacuum System Parameters

Component	Dimensions (cm)	Conductance (m^3/s)
Limiter slots (2)	5650 × 10 × 50	4300
Limiter ducts (2)	3170 × 16 × 70	4100
Plenum	6000 × 67 × 600	13700
Vacuum ducts (24)	100 × 640 120 × 560	730
Vacuum pumps (24)	——	2900

Rated helium speed per pump, m^3/s	120
Rated DT speed per pump, m^3/s	200
Total helium pumping speed, m^3/s	490
Transmission probability[a] (helium)	0.9
Reflection coefficient (helium), R_α	0.75
Maximum helium pressure, Pa	0.016
Total DT pumping speed, m^3/s	480
Transmission probability[a] (DT)	0.40
Reflection coefficient (DT), R_{DT}	0.9
Maximum DT pressure, Pa	0.024
Tritium fractional burnup	0.42
Total gas load, Pa-m^3/s	18.7
DT gas load, Pa-m^3/s	10.85
Helium gas load, Pa-m^3/s	7.85
Temperature, °K	573
Number of vacuum pumps, on-line/total	24/48
Regeneration time, h	2
Maximum tritium inventory per pump, g	2.6

[a]Transmission probability per particle entering the limiter slot.

tritium inventory in the vacuum pumps and fueling system.

The STARFIRE study finds it an important design approach to radiate most of the alpha-power from the plasma to the large surface area of the first wall. This reduces the heat load on the particle collection medium (limiter or divertor target plate) to a manageable level and it deposits more energy in the primary coolant of the first wall. One means of enhancing plasma radiation is by injecting small amounts of high-Z material along with the DT fuel stream. The large ignition margin in commercial reactor-size plasmas makes operation in such an enhanced radiation mode feasible.

A low-Z coating on all surfaces exposed to the plasma will probably be required in future tokamak reactors unless very low plasma edge temperatures can be established and maintained. Beryllium appears to be one of the best choices for the low-Z coating. Sputtering of the limiter coating is predicted to be large but redeposition seems to extend the coating life to an acceptable level. However, there is a need for experimental results and theoretical work on the physics of the scrape-off region and the performance of low-Z coatings. There is also a need to develop in-situ low-Z coatings techniques for fusion reactor applications.

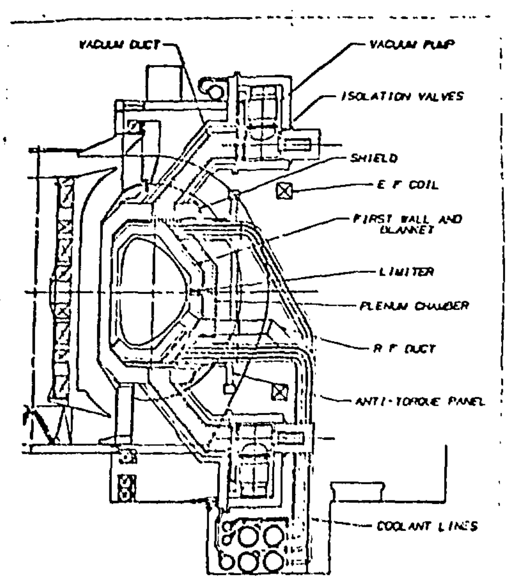

Fig. 3. A cross section of the STARFIRE reactor showing the components of the vacuum system.

Four materials have been identified as the most promising for the limiter structure. These are alloys of copper (AMAX-MZC), tantalum (Ta-5W), niobium (FS-85) and vanadium (V-20Ti). These alloys can withstand the high heat fluxes on the limiter. Available data indicates that these alloys have many properties that are suitable for the reactor environment. Unfortunately, the data base is not complete and more information is required in the areas of corrosion and radiation effects for all of these alloys.

The results of STARFIRE indicate that a high efficiency exhaust system is not necessarily desirable. It is very beneficial to keep the removal efficiency low so that the tritium fractional burnup is high. This reduces the gas load in the exhaust system and simplified the vacuum system design in addition to lowering the vulnerable tritium inventory in the fueling and vacuum systems.

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