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THE CHOICE OF COOLANT IN COMMERCIAL TOKAMAK POWER PLANTS*

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Summary

The STARFIRE design study focused on solid tritium breeder blankets in order to minimize the stored chemical energy. The most suitable coolant candidates with solid tritium breeders are water and helium. This paper presents the results of a comparative study of the two coolants. The study shows clear advantages for the choice of pressurized water for the conditions of the STARFIRE tokamak power plant design. The study also identifies those areas where development is required in order to utilize the potential advantages of helium.

1. Introduction

The choices of coolant and tritium breeding material have a substantial impact on the design, operation, maintenance, safety, and economics of a fusion power plant. Therefore, a great deal of attention was devoted in the STARFIRE[1] study to the choices of coolant and breeder material. The solid tritium breeder, $LiAlO_2$, and pressurized water coolant were selected for the reference design. The rationale for this selection is presented in this paper.

The promising coolant types are liquid metals, molten salts, helium, and water. Liquid lithium offers unique advantages. It can simultaneously perform the functions of tritium production, heat deposition, and heat transport resulting in a simple low-pressure system. It is also compatible with most structural materials. The major problem with liquid lithium is the large stored chemical energy. The associated safety problems are of concern. Previous design studies[2-4] and experience from the LMFBR program indicate that special design features, e.g. multiple barriers between liquid lithium and air and water, can reduce the probability of lithium fires to very low levels. However, these preventive design measures are costly. Furthermore, liquid lithium has other disadvantages that include difficult problems in maintenance and cleanup of spills, a need for an intermediate coolant, and MHD effects. Therefore, it seems prudent at this stage of fusion research and development to seriously explore other options t .c offer intrinsic safety features. It was in this spirit that a decision was made in the STARFIRE study to fucus on solid breeders. No detailed comparison between liquid lithium systems and the reference solid breeder/water system was attempted. Such a study should be performed in the future.

The major solid breeder candidates are Li_7Pb_2, Li_2O, and the ternary oxides such as $LiAlO_2$, Li_2TiO_3, and Li_2SiO_3. Important criteria considered in the selection of potentially viable solid breeding materials include chemical stability, compatibility, neutronics properties, and tritium release characteristics. The α-$LiAlO_2$ is selected on the basis of the best combination of these materials requirements.[5,6]

The allowable operating temperature ranges for the candidate compounds have been predicted from available thermodynamic data. The low-temperature limits, which are defined by tritium diffusion kinetics in the solid, are based on very small ($\sim$ 1 μm) grain size. The upper temperature limits are based on sintering characteristics of the solids which would close interconnected porosity and increase the diffusion path. Allowances for radiation-induced trapping of tritium at the lower temperatures and radiation-induced sintering at the higher temperatures are included.

The ceramics are preferred over the intermetallic compounds for the reference solid breeding material because of the larger allowable operating temperature ranges. On the basis of this criterion, Li_2O and $LiAlO_2$ appear to have an advantage. However, the calculated solubility of tritium in Li_2O at these temperatures and at reasonable T_2O partial pressures in the tritium-processing stream ($> 10^{-1}$ Pa) is much greater.[6] Also, compatibility of Li_2O with the structural material is a major concern.

Li_7Pb_2 was excluded on the basis of its reactivity with the water coolant and compatibility problems with the structural material at the high temperatures required for helium cooling.

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The most suitable coolant candidates with solid tritium breeders are water and helium. The water coolant can be D_2O or H_2O either as steam or liquid.

Heavy water has several advantages compared to H_2O. The requirements on the processing of tritium from the water coolant is less difficult for D_2O since both deuterium and tritium are used for the plasma fuel. Another important advantage of D_2O relates to its neutronics properties. D_2O has a lower neutron slowing down power and a smaller neutron capture cross section than H_2O. Therefore, D_2O has less negative impact on the tritium breeding ratio and it gives more uniform lithium burn-up and energy deposition. However, the very high cost of D_2O makes it economically undesirable. Steam was given some consideration but was found to offer no overall advantages over pressurized water or helium.

The choice between helium and pressurized water (H_2O) was a difficult decision involving several issues. This difficulty can be appreciated by examining Table 1 which shows a large number of technical areas of the reactor design affected by the choice of one of these two coolants. Therefore, a comparative study of helium and pressurized light water was performed. The early comparison was performed using the ANL systems code[7] and showed clear benefits for the choice of pressurized water for the typical conditions of STARFIRE. After the STARFIRE design was completed, the comparative study was repeated on "a point basis". Starting with the reference pressurized water system, the areas most affected by the choice of coolant were examined and the performance characteristics and costs for a helium system were estimated and compared to the reference water design. The results of this point comparison confirmed the prudence of the earlier decision to use water. Key points from this point design comparison are discussed below.

2. Gross Thermal Efficiency

In reference to Table 1, the first area of comparison is the thermal conversion efficiency. Helium cooling is an advanced technology with potentially higher conversion efficiency than pressurized water. The reason for this is that a helium gas can be operated at high temperatures (e.g. 700–900°C) with reasonably low pressures (e.g. 700–1000 psi) while water requires high pressures at low temperatures (critical temperature ∿ 370°C). However, a key problem that must be clearly realized is that there is no structural material identified at present that (a) can operate at high temperatures; (b) is resistant to radiation damage; and (c) is compatible with impurities in helium. Structural material temperatures < 500°C are not capable of utilizing the full potential of helium. Modified austenitic stainless steel, PCA, is used in STARFIRE with water and is judged to be the only acceptable material with helium if no aggressive materials development program is pursued. This is a key assumption in this comparison.

The problems of cooling the first wall with helium are particularly severe. The heat deposited in the STARFIRE first wall is ∿ 20% of the reactor thermal power and the surface heat load on the wall is high, ∿ 0.9 MW/m^2. Attempts to design a helium-cooled first wall resulted in large pumping power requirements. Therefore, it is assumed in this study that for the helium case, the first wall is cooled with water while the helium is used only for cooling the blanket. The helium coolant exit temperature is 475°C and the water from the first wall is used for feedwater heating in the steam turbine cycle. The maximum coolant temperature for the reference pressurized water case is 320°C.

The obtainable thermodynamic efficiency depends on the steam temperature which in turn depends on the pinch-point temperature difference in the steam generator. To keep the cost of the steam generator reasonable, a pinch point of ∿ 10°C is normally maintained with water and ∿ 30 – 100°C with helium. The higher pinch point necessary with helium is due to its poorer heat transport properties. The gross thermal conversion efficiency with water is calculated as 35.7%, based on a steam temperature of 299°C (570°F) and a pressure of 6.3 MPa (910 psi) and assuming minor improvement in steam turbine technology. A steam temperature of 427°C (800°F) and pressure of 13.1 MAa (1900 psi) are assumed with helium. The resulting pinch-point in this case is only 13°C which increases the size of the steam generator. The gross thermal conversion efficiency for the helium case is taken to be 40%. Since 20% of the thermal power is removed by water from the first wall and 200 MW of heat is removed by low temperature water from the limiter, the assumed conversion efficiency for helium alone is > 40%. Optimistic conditions are assumed here for helium. A comparison of helium performance assumed in this study with that in present gas-cooled reactors[8] is shown in Table 2.

3. Pumping Power

Another major difference between the two coolants is the pumping power requirements. These are low with water, ∿ 0.3–0.8% of the thermal power. The pumping power for helium is generally large and is roughly inversely proportional to the square of the helium pressure and coolant temperature rise. The operating pressure in current gas-cooled reactors (see Table 2) is ∿ 700 psia. However, studies for recent gas-cooled fast reactor (GCFR) designs show significant economic incentives for technology development to accommodate high helium coolant pressures.[9] A pressure of 1500 psia is assumed for helium in this comparative study. Considerations of tritium release from solid breeders necessitates a small coolant temperature rise, ΔT = 40°C. Such a small ΔT would result in very large pumping power for helium. Therefore, a ΔT of 175°C is assumed for the helium case. (This will result in a higher blanket tritium inventory.) Under these conditions, the pumping

Table 1. Key Areas of Comparison Between Helium and Water Coolants in STARFIRE

1. Gross thermal efficiency
2. Pumping power requirements
3. Inner blanket, shield thickness (void space with helium)
4. Primary coolant loop plus steam generator cost
5. Outer blanket/shield thickness
   — Outer leg of the TF coil
   — Design of EF coils
6. Shielding
   — Streaming through helium ducts
   — Crud in water loops
7. Achievable tritium breeding ratio in solid breeders
8. Ease of in-situ tritium recovery from solid breeders with narrow temperature range
9. Chemical reactivity
10. Coolant leakage
11. Overpressure on reactor building
12. Requirements for an intermediate coolant loop
13. Cost of coolant and coolant purification (e.g. crud and tritium)

Table 2. Comparison of Typical Coolant Conditions in Present Gas-Cooled Reactors with Conditions Assumed in This Study

	Fort St. Vrain HTGR DEMO	Peach Bottom HTGR Prototype	Conditions Assumed in the Present Study[a]
Reactor coolant exit temperature (helium inlet to steam generator), °C	776	734	475
Steam temperature, °C	538	538	427
Gross thermal efficiency, %	40.7	39.9	40[b]
Steam generator pinch point, °C	146	98	13
Coolant pressure, psi	700	350	1500
Steam pressure, psi	2400	1450	1900
Temperature rise of helium in reactor, °C	372	400	175

[a]For the purpose of comparing helium coolant performance to the reference pressurized water coolant in STARFIRE.

[b]Assuming roughly a 2-point improvement in thermal efficiency due to advances in steam turbine technology.

power requirements for helium are ~ 6% of the thermal power transported by helium. Since 20% of the thermal power is removed by water in the first wall, the pumping power for the helium case is ~ 4.7% of the reactor thermal power. The fact that this pumping power is added to the circulating fluid as thermal energy and converted to electric power must also be accounted for.

4. Inner Blanket/Shield Thickness

In STARFIRE, the use of a solid breeder is an important feature. All useful solid breeders (ternary oxides) that satisfy the tritium recovery and material compatibility constraints require a neutron multiplier and have much lower tritium breeding potential than liquid lithium. The relatively large percentage of the structural material required with the helium coolant makes development of blanket designs with a reasonably conservative margin in the tritium breeding ratio very difficult. It was concluded[1] from the neutronics analysis that a blanket breeding region must be placed on the inner side of the torus. This conclusion strongly impacts the helium/water comparison in view of the negative effect of void space in the inner blanket on tokamak reactor performance and economics. For a given plasma geometry, beta, and maximum toroidal field, the fusion power varies with the inner blanket/shield thickness, Δ_{BS}^{i}, as

$$P_f \sim \left(\frac{R - a - \Delta_v - \Delta_{BS}^{i}}{R}\right)^4 ,$$

where R is the major radius, a is the plasma half-width, and Δ_v is the scrape-off thickness. For STARFIRE, R = 7 m, a = 1.94 m, and Δ_v = 0.2 m. The required blanket/shield thickness with water coolant is Δ_{BS}^{i} = 1.2 m.

Helium requires $\Delta_{BS}^{i} = 1.2 + \delta$, where δ is the equivalent thickness of the void space for the helium coolant in the inner blanket. Solid breeder blanket module designs were developed in sufficient detail to permit reasonable estimates of δ. A typical value for δ was found to be ~ 0.38 m with 0.18 m void in the blanket region and 0.2 m equivalent void space for manifolds and headers. The void space can be reduced by increasing the helium pressure. Furthermore, clever routing of the coolant manifolds and locating the headers further away from the midplane in the inboard blanket will substantially reduce δ, but will also significantly increase the coolant pumping power requirements. In keeping with the spirit of this comparison of cinsidering an optimistic case for helium, the value of δ is taken as only 0.18 m; thus Δ_{BS}^{i} for the helium case is 1.38 m. This reduces the thermal power for the helium case by ~ 18%.

5. Cost of Heat Transport System

Tables 3 and 4 show the heat transport parameters for water (reference STARFIRE case, see Ref. 1) and helium, respectively. The water system

Table 3. Water Heat Transport System Parameters

Coolant	Water
Heat load	3800 MW
Water temperature	
Reactor outlet	320°C
Reactor inlet	280°C
Operating pressure	2200 psi
No. of independent loops	2
Maximum pipe size (i.d.)	0.99 m
Maximum velocity	20 m/s
Gross system pumping power	30 MW
No. of pumps	4
No. of steam generators	4
Steam temperature	299°C (570°F)
Steam pressure	6.3 MPa (910 psi)

Table 4. Helium Heat Transport System Parameters

Coolant	Helium
Heat load (total)	3105 MW
Blanket (helium)	2484 MW
First wall (water)	621 MW
Helium temperature	
Blanket outlet	475°C
Blanket inlet	300°C
Operating pressure	1500 psi
No. of loops	4
Maximum pipe size (i.d.)	1.13 m
Maximum velocity	103 m/s
Gross system pumping power	150 MW
No. of helium circulators	4
No. of steam generators	4
Steam temperature	427°C (800°F)
Steam pressure	13.1 MPa (1900 psi)

utilizes two loops while the helium system incorporates four parallel loops beyond the dual ring manifold system. Each water loop utilizes two steam generators and two pumps while only one steam generator and one helium circulator are used in each helium loop.

Component costs for the two systems were estimated as shown in Table 5. The costs for the water system were taken from the reference STARFIRE case.[1] The costs for the helium case were estimated by using unit costs or by scaling from the more detailed cost estimates of HTGR and GCFR. The hot leg of the helium piping incorporates a thermal barrier as in the HTGR designs. As shown in Table 5, most components in the gas-cooled system are more expensive, reflecting the larger volumetric flow rate of coolant to be handled. On a per unit thermal power basis, the helium heat transport system cost is almost twice that of a water-cooled system.

6. Other Areas of Comparison

The above discussion addressed only the first four areas of comparison given in Table 1. It is in these four areas that major differences in the performance and economics between water and helium are found. The other areas are important but it is not clear at present that any one of them will make so large a difference as to change or overwhelmingly strengthen the impact of the first four areas. A brief discussion of these other areas follows.

Table 5. Comparison of Cost Estimates for the Heat Transport System of Helium and Water-Cooled Systems

Component	Cost Estimate (millions $)	
	Water Cooled	Helium[a] Cooled
Piping and manifolds	26.1	35.5
Valves	6.1	5.9
Pumps/circulators	2.4	14.0
Pressurizers	6.3	(b)
Coolant makeup and cleanup	4.2	5.5
Steam generators	18.0	40.0
Additional components for water-cooled first wall	(b)	10.2
TOTAL	63.1	111.1

[a]Helium system incorporates water-cooled first wall dissipating 20% of the power.

[b]Not applicable.

One difference between helium and water is the outer blanket/shield thickness, Δ_{BS}^{o}. As discussed earlier for the inboard region, Δ_{BS}^{o} will have to be increased with helium. This results in an increase in the major radius of the outer leg of the TF coil, thereby increasing the cost of the TF and the externally located equilibrium field (EF) coils.

Another difference between the two coolants is the shielding requirements. The large-size helium manifolds will result in significant radiation streaming. Adequate shield has to be provided around the helium manifolds. In addition to the cost of the shield, the large-size manifolds and the shields that surround them complicate the space problem between the outer legs of the TF coils. On the other hand, some of the corrosion products with water will be transported throughout the heat transport system. Since these corrosion products are radioactive, shielding of the heat transport pipes for the water coolant is necessary if personnel access into the reactor building after shutdown is required.

Helium-cooled blankets require a larger amount of structural material than that for water. As indicated earlier, the most promising solid breeders (ternary oxides, e. g. $LiAlO_2$) have low tritium breeding potential. The increase in parasitic absorption in the structural material makes it difficult in most cases to attain a tritium breeding ratio greater than one with helium cooling. This problem can be resolved if it is shown that the tritium release characteristics of Li_2O are substantially better than currently predicted or if the compatibility problems associated with Li_7Pb_2 can be solved.

In order to keep the tritium inventory in the blanket low, solid breeders have to operate in a narrow temperature range. As discussed earlier, a low coolant temperature rise in the blanket makes the pumping power with helium unacceptably large. It appears therefore, that a high tritium inventory in the blanket may be unavoidable with helium cooling.

A disadvantage of water is its reactivity with lithium and some of the lithium compounds. A major advantage of helium is that it is chemically inert. The chemical reactivity of water with the ternary oxides of lithium is generally weak.

Helium leakage is of concern. It is estimated that the entire inventory of the helium coolant is lost roughly once every year due to leakage. This problem is compounded by the use of a high helium pressure to minimize the coolant pumping power.

The maximum overpressure for which the reactor building is designed depends to some extent on the coolant pressure. Helium shows an advantage here as its operating pressure is lower than that for water. However, in designs such as STARFIRE, the thickness of the reactor building wall is determined more by the biological shielding requirements[11] rather than overpressure considerations.

The STARFIRE study has shown that an intermediate coolant loop between the primary water coolant and the steam cycle is not necessary to keep the tritium leakage to the environment low. This is expected to be also the case for helium cooling. If, for other presently unknown reasons, either of the two coolants is shown to require an intermediate coolant loop, this can result in a significant impact on the present comparison.

Water represents a sink for tritium. While this has the favorable effect of preventing tritium permeation to other regions of the reactor, it is a cause for concern in cases of accidents involving release of water. This problem can be circumvented by keeping the tritium concentration in the water coolant sufficiently low. As discussed in Ref. 12, this requires continuous or periodic processing of water to remove tritium. However, the cost for this capability is estimated to be small relative to the economic impact of the first four major areas of Table 1.

7. Power Plant Economics and Conclusions

Table 6 shows the key points regarding the comparison of performance and economics for pressurized water and helium coolants. A dominant effect is the increase in the inner blanket/shield thickness when the helium coolant is used. For the same maximum magnetic field (11.1 T), plasma average beta (0.067) and the same reactor plasma size (major radius = 7 m, minor radius = 1.94 m), the fusion power in the helium case is ~ 18% lower than in the reference water case. This effect, together with the larger helium pumping requirements, cause the net electric power to be lower with helium despite its larger thermal conversion efficiency. The cost of the heat transport system is considerably higher for helium. The reference power plant costs for STARFIRE were modified for the helium case to reflect reductions in the costs of the turbine plant equipment, the electric plant equipment, and other accounts. These reductions are due to the lower electric power generation with helium.

Table 6. Summary of Key Points in the Water and Helium Coolants Comparison for STARFIRE

	Water	Helium
Pressure, psi	2200	1500
Inner blanket/shield thickness, m	1.20	1.38
Maximum magnetic field, T	11.1	11.1
Magnetic field on axis, T	5.80	5.52
Thermal power, MW	4000[a]	3305[a,b]
Coolant temperature, °C		
Reactor exit	320	475
Reactor inlet	280	300
Gross thermal efficiency, %	35.7	40
Coolant pumping power, MW	33	153
Other auxiliary power, MW	207	207
Net electric power,[c] MW	1200	1011
Cost of primary coolant loop	63	111
Direct plant capital cost, M$	1700	1620
$/kWe (relative units)	1.0	1.13

[a] Includes 200 MW in the limiter, which is removed by a separate water coolant.

[b] 621 MW of this power is deposited in the first wall and is removed by a water coolant.

[c] Account is made for the fact that the pumping power is added to the coolant as thermal energy; efficiency of helium circulator = 0.8.

The cost per unit power is shown to be significantly higher, ∿ 13%, with helium cooling. Therefore, the choice of pressurized light water over helium for STARFIRE is clearly justified.

Helium has many attractive features but it is not economically competitive if constaints based on the present state-of-the-art are imposed. However, it is strongly emphasized here that helium cooling cannot be ruled out now as an option for fusion reactors. Rather, it is hoped that the present results can help identify the technology development requirements that are necessary for effective utilization of the helium cooling option. One of the most important of these requirements appears to be the development of high temperature (> 600°C) structural materials that are compatible with impurities in helium. For example, there seems to be a great incentive for attempting to resolve the compatibility problems between the vanadium alloys (low long-term radioactivity and good resistance to radiation damage) and the achievable minimum level of impurities in helium. Another important area is resolving the problem of void space in the inner blanket. A key to this is the use of a breeding material whose tritium breeding capability is large enough so that placing a breeding module in the inboard blanket is not necessary. The only breeding materials that offer this potential are natural lithium, some of the lithium-lead-bismuth compounds, and Li_2O.

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