ENERGY CONVERSION CONSIDERATIONS OF THE STARFIRE COMMERCIAL FUSION POWER PLANT*

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Abstract

STARFIRE is a conceptual design for a commercial tokamak power plant based on the deuterium/tritium/lithium fuel cycle. STARFIRE operates in a steady state mode with the plasma current driven by lower hybrid rf. The plasma impurity control and exhaust system is based on the limiter/vacuum concept. The reactor has a 7-m major radius and produces 4000 MW of thermal power with an average neutron wall load of 3.6 MW/m^2. The first wall/blanket structure is PCA stainless steel. A solid neutron multiplier (Zr_5Pb_3) and a solid tritium breeder ($LiAlO_2$) are utilized. The primary coolant is pressurized water (15.2 MPa) with inlet and outlet temperatures of 280°C and 320°C, respectively.

1.0 Introduction

The basic purpose of the STARFIRE study[1] is to develop a design concept for a commercial tokamak fusion electric power plant based on the deuterium/tritium/lithium-fuel cycle. This paper summarizes the major features of the reference reactor concept and describes in detail the energy conversion systems of the reactor and power plant. The basic design guidelines for STARFIRE assume the successful operation of a tokamak engineering test facility and a demonstration power plant. STARFIRE is considered to be the tenth plant in a series of commercial reactors. It is, therefore, assumed that a well-established vendor industry exists and that utilities have gained experience with the operation of fusion plants.

Safety has played a major role in considering various blanket options. Solid tritium breeders instead of liquid lithium have been emphasized in this study. In addition, efforts have been made to minimize the tritium inventory in the plasma exhaust processing systems and the radioactivity induced in the materials in the magnets and shield.

The major features for STARFIRE include a steady-state operating mode based on a continuous rf lower-hybrid current drive and auxiliary heating, solid tritium breeder material with no liquid lithium, pressurized water cooling, limiter/vacuum system for impurity control, all superconducting EF coils outside the TF superconducting coils, fully remote maintenance, and a low-activation shield. These features have resulted in a simplified tokamak reactor concept while increasing the attractiveness of the reactor with respect to safety and environmental features.

Availability goals have been established as 85% for the reactor and 75% for the complete plant including the reactor. These goals provide a basis for design of maintenance equipment. The maintenance scenario incorporates the current utility practice of shutting down annually for one month and a four-month shutdown approximately every five to ten years.

2.0 Overview of the STARFIRE Concept

The major parameters of STARFIRE are summarized in Table 1. An isometric view of the reference design is shown in Fig. 1. This section provides an overview of the reactor concept. The reactor and plant energy conversion systems are described in more detail in Sec. 3 and 4, respectively.

Past and ongoing research in plasma physics indicates the possibility that toroidal plasma currents may be maintained in tokamaks with noninductive external momentum sources to the electrons. This suggests that steady state may be an achievable mode of operation for tokamaks. Steady-state operation offers many technological and engineering benefits in commercial reactors. Among these are that component and system reliability is increased, material fatigue is eliminated as a serious concern, higher neutron wall loads are acceptable, thermal energy storage is not required, the need for an intermediate coolant loop is reduced, electrical energy storage is significantly reduced or eliminated, a full-size ohmic heating solenoid is not needed, and external placement of the EF coils is simplified.

The lower hybrid wave has received the most extensive study for current drive in tokamaks, and on this basis it was selected for the STARFIRE design. A plasma equilibrium was selected with a total current of only 10 MA. For the aspect ratio ($A = 3.6$) and elongation ($\kappa = 1.6$) characterizing the D-shaped plasma cross section of STARFIRE, the plasma was found to be stable to at least the design value of $\beta_t = 0.067$ against interchange, ballooning, and low-mode number kink plasma MHD instabilities. For STARFIRE, the antenna must deliver 63 MW at 1.4 GHz. Enough power is lost to the sidebands that a total of 90 MW must be absorbed in the plasma. The total electrical power requirement for the rf system is 150 MW.

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Net electrical power, MW	1200
Gross electrical power, MW	1440
Fusion power, MW	3490
Thermal power, MW	4000
Gross turbine cycle efficiency, %	36
Overall availability, %	75
Average neutron wall load, MW/m^2	3.6
Major radius, m	7.0
Plasma half-width, m	1.94
Plasma elongation (b/a)	1.6
Plasma current, MA	10.1
Average toroidal beta	0.067
Toroidal field on axis, T	5.8
Maximum toroidal field, T	11.1
No. of TF coils	12
TF coils material	Nb_3Sn/NbTi/Cu/SS
Blanket structural material	Austenitic steel (modified, PCA)
Tritium breeding medium	α-$LiAlO_2$
Neutron multiplier	Zr_5Pb_3 (solid) or Be
First wall/blanket coolant	Pressurized H_2O

breaking down the ... ∿ 5 MW electron cyclotron resonant heating (ECRH) system, inducing 1-2 MA of plasma current with OH coils and building up and sustaining the 10-MA plasma current using an rf system. Plasma fueling is accomplished via gas puffing or possibly pellet or plasma gun injection.

An important design consideration is the choice of the plasma impurity and alpha-particle removal concept. Investigations in this study indicate that modest pumping of helium with a limiter/pumping system (∿ 25% of the alpha-particle flux) coupled with about a 1.5-T margin in the maximum toroidal field should eliminate the need for a divertor. This result is based on the provision that a significant portion of the alpha-particle heating power can be radiated to the first wall rather than be deposited on the limiter. In general, a non-divertor option is greatly preferred from an overall reactor engineering point of view.

During plasma operation the plasma impurities, including alpha particles, are removed using a limiter system and continuous vacuum pumping. The limiter consists of tantalum segments which form a continuous toroidal ring at the reactor outer midplane. The limiter is subjected to a peak heat flux of 4 MW/m^2 and is cooled with 150°C water which is used for feedwater heating. As particles impinge on the limiter, ∿ 25% are directed into a slot behind the limiter. These particles are then pumped through a vacuum plenum region between the blanket and shield into 24 vacuum ducts at the top and bottom of the reactor. Forty-eight cryosorption/cryocondensation pumps are used. Twenty-four of the pumps are operated while the remaining twenty-four are rejuvenated. Pumps are rejuvenated hourly to minimize tritium inventory.

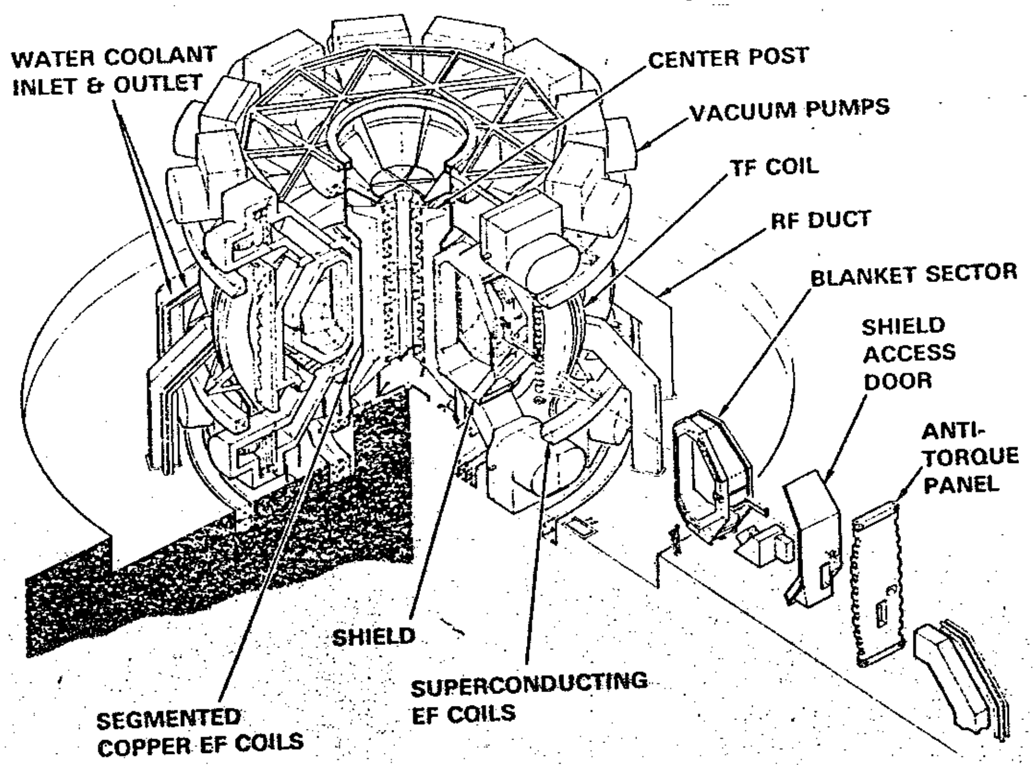

Fig. 1. STARFIRE reference design – isometric view.

The limiter/vacuum system achieves a very high fuel utilization efficiency, having a tritium fraction burnup of 35%. As a result, tritium flow rates in the fuel cycle are very low, about 1 kg/day or 250 g/GWth-day. By comparison, most previous reactor designs have a much lower fractional burnup, 1-10%, and the corresponding fuel cycles have to process 1-10 kg/GWth-day. The tritium inventory considered to be vulnerable to accidental release is < 400 grams.

Another key design consideration is the location of the equilibrium field (EF) coils. The basic design approach is to locate almost all the EF coils outside of the toroidal field (TF) coils. All such EF coils would be superconducting. A limited number of segmented copper coils are located inside the TF coils, but outside of the blanket and shield.

The reactor magnet system consists of 12 TF coils, 8 superconducting EF coils, 4 normal conducting EF coils, and 6 OH coils. Two of the OH coils are combined with the EF coils to simplify assembly. No intertwined superconductors are used and the normal coils are segmented to permit maintenance. The TF coils have a common vacuum dewar at the inner coil leg and separate vacuum dewars on the outer leg. TF coil overturning moments are reacted through the 4°K center post and room temperature shear panels between adjacent TF coil outer legs. The room temperature TF coil case also supports the EF coils and the shield assembly. The vacuum pumps utilize an additional support frame.

The first wall is an integral part of the blanket structure (see Sec. 3.0). The blanket is segmented toroidally into 24 sectors to permit removal between TF coils. Two different sector sizes are used to permit location of the high-pressure coolant line disconnects outside the vacuum chamber. The first-wall and structural material is PCA stainless steel that operates at $\sim$ 425°C maximum temperature when subjected to an average neutron wall load of 3.6 MW/m^2. The first-wall blanket is cooled by water with inlet and outlet temperatures at 280°C and 320°C, respectively. This permits operation of $LiAlO_2$ solid breeder material within a proper temperature range to enhance tritium release without sintering. A helium purge stream is used to extract the tritium.

The first-wall/blanket sectors also provide mounting for the 12 ECRH and 12 lower-hybrid waveguides, the fueling ports, and the limiter system. The waveguides and fueling ports are located on the sector between TF coils. The first wall, limiter, and waveguides are coated with beryllium to minimize the effects of sputtered impurities on the plasma. The first wall/blanket, limiter, and waveguide assembly are designed for a 16 $MW\text{-}yr/m^2$ life. Blanket sectors are manifolded separately to permit leak detection and isolation.

The shield provides neutron and gamma-ray attenuation and serves as the primary vacuum boundary for the plasma. The outer shield is composed of titanium, lead, B_4C and H_2O, which offers significant environmental advantages with respect to minimum radioactive waste considerations. The shield is assembled from 12 sectors and 12 shield rings. Dielectric breaks are located in six of the shield rings near the outer surface of the shield to limit the radiation dose to 10^{10} rads. The Kapton dielectric seal is factory installed and designed for life-of-plant operation. Removable shield doors[2] are located between TF coils to permit blanket sector removal. The shielding is $\sim 10^8$ rads so that elastomer door seals can be used. Redundant seals and dielectric breaks are used to permit leak detection and isolation.

A steam power conversion system (Sec. 4.0) without an intermediate heat exchanger is utilized to convert the reactor thermal energy to electrical power. Two separate heat removal circuits are utilized, one for the first wall/blanket and the other for the limiter. The power deposited in the limiter (200 MW) is used for feedwater heating while the recoverable power (3800 MW) from the first wall and blanket is used to produce steam at 299°C and 6.3 MPa. The steam is then used in a turbine-generator unit for producing 1440 MW of electric power. The net electrical power is 1200 MW with 240 MW recirculating power for the rf system, coolant pumps, and other reactor subsystems.

3.0 Primary Energy Conversion System

The 4000 MW of thermal power in the STARFIRE power plant is recovered from two components. The first wall/blanket component produces 3800 MW, which is transported into the steam generator by pressurized water at 320°C. Approximately 200 MW is produced in the limiter and is removed by a low pressure ($\sim$ 2.8 MPa) water coolant at 165°C. This low temperature heat is utilized for feed-water heating in the turbine cycle. In addition to the 4000 MW of recoverable energy, low-grade heat (< 100°C) is produced in other reactor components such as the shield and lower-hybrid rf grills.

This section presents the key aspects of the primary energy conversion system in STARFIRE. Section 3.1 is devoted to the first wall and blanket while Sec. 3.2 describes the limiter component.

3.1 First Wall/Blanket

The selection of the first wall/blanket design for STARFIRE evolved from a detailed examination of promising material options and design concepts. Major emphasis has been placed on safety and environmental acceptability, with primary goals that include minimal stored chemical energy and low tritium inventory in the blanket. The primary focus in the study has been on concepts that utilize solid lithium compounds for the tritium breeding material.

3.1.1 Neutron Wall Load

A key parameter in characterizing the operational environment for the first wall and blanket is the neutron wall load. Another very important and related parameter is the surface heat load on the first wall.

The maximum value of the surface heat load is 25% of the neutron wall load. In the presence of a plasma impurity control and exhaust system, the surface heat load can be significantly reduced as the plasma radiation decreases and the charged particles diffusing out of the plasma are diverted away from the first wall. While this reduces the surface heat load, and hence thermal stresses, on the first wall, the heat load becomes intolerably high on the small surface area of the collector plate of the divertor or limiter. The STARFIRE strategy is to radiate most of the alpha-heating power to the large surface area of the first wall by injecting a small amount of Xenon along with the deuterium-tritium fuel stream. Since STARFIRE is operated in a steady-state mode, a moderately-high surface heat load on the first wall can be tolerated.

A higher neutron wall load permits a smaller size reactor and lower capital cost. On the other hand, there are limitations on the maximum neutron wall load. There are constraints on the maximum power density producible in the plasma. Furthermore, important considerations in the design of the first wall, such as the maximum structural material temperature, structure lifetime and coolant pumping power limit the maximum allowable neutron wall load. Trade-off studies[3] to minimize the cost of energy in STARFIRE resulted in selecting an average neutron wall load of 3.6 MW/m^2. These results assumed 16 $MW\text{-}yr/m^2$ for the stainless steel structure and a total cumulative downtime for replacement of the first wall and blanket structure of 125 days. The average surface heat load is 0.9 MW/m^2. A peak-to-average ratio of $\sim$ 1.2 is estimated for the reference design.

3.1.2 Choice of Coolant

The choice of coolant has a substantial impact on the design, operation, maintenance, safety, and economics of a fusion power plant. The promising coolant types are liquid metals, molten salts, helium, and water. Liquid lithium offers unique advantages.[4] 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. However, the potential safety problems associated with the relatively large stored chemical energy in liquid lithium systems provide an incentive for seriously examining other options. A promising system is a non-mobile solid lithium compound blanket with helium, water, or a molten salt as a coolant. The advantage of low operating pressure with molten salts is outweighed by the disadvantages of higher melting temperature, incompatibility with many structural materials and induced radioactivity problems. Therefore, helium and water are the only two attractive candidates for cooling solid-breeder blankets. A comparative study[5] of the two coolants was performed. The study proved to be rather complex as there are approximately 15 technical areas of the reactor affected by the coolant choice. Nevertheless, the study showed clear advantages for the choice of water for the conditions of STARFIRE. Key points from the study are summarized below.

Helium cooling is an advanced technology with potentially higher conversion efficiency than pressurized water. However, a key problem that must be clearly recognized is that there is no structural material identified at present that can operate at high temperature, is compatible with practical levels of impurities in helium, and has good resistance to radiation damage. Structural material temperatures < 500°C are not capable of utilizing the full potential of helium. With modified austenitic stainless steel, the maximum coolant exit temperature is 475°C with helium and 320°C with water. 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 $\sim$ 10°C is normally maintained with water and $\sim$ 50-100°C with helium. The gross thermodynamic efficiency, η, is in the range of 36-39% for helium coolant and 34-36% for pressurized water.

Another major difference between the two coolants is the pumping power requirements. These are low with pressurized water, typically $\sim$ 0.3% of the thermal power. The pumping power for helium is generally large and is approximately inversely proportional to the square of the helium pressure and coolant temperature rise. Helium coolant pressures much in excess of 1000 psi require extrapolation in technology. The magnitude of the coolant temperature rise is severely limited by the constraint on the maximum coolant exit temperature discussed above. It was estimated that the pumping power for helium at 1500 psi is $\sim$ 3% of the thermal power.

In STARFIRE, stipulation of a solid breeder blanket is an important feature. All useful solid breeders that satisfy the tritium recovery and material compatibility constraints (with the possible exception of Li_2O) 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 does not permit development of blanket designs with a reasonably conservative margin in the tritium breeding ratio. It was concluded 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^i_{BS}, as

$$P_f \propto \left(\frac{R-a-\Delta_v-\Delta^i_{BS}}{R}\right)^4 ,$$

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^i_{BS} = 1.2 m.

Helium requires $\Delta^i_{BS} = 1.2 + \delta$, 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. 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 substantially reduces δ, but also significantly increases the coolant pumping power requirements. The minimum value obtained in acceptable designs is δ = 0.18 m.

Table 2 shows a comparison of STARFIRE performance with water and helium coolants. The void space thickness with helium cooling is taken as 0.18 m and has a substantial impact on the results of the comparison. For the conditions considered here, the net electrical power output is 1200 and 985 MW with pressurized water and helium, respectively. The cost of the primary coolant loops (pipes, pumps, and steam generators) is much more

Table 2. Summary of Key Points in the Water and Helium Coolants Comparison for STARFIRE

(Reference parameters: R = 7 m, a = 1.94 m, Δ_v = 0.2 m, B_m = 11.1 T, β_t = 0.067)

	Water	Helium
Pressure, psi	2200	1500
Δ^i_{BS}, m	1.20	1.38
B_o, T	5.80	5.52
Thermal power, MW	4000	3273
Coolant exit temperature, °C	320	475
Gross thermal efficiency, %	36	39
Gross electric power, MW	1440	1305
Coolant pumping power, MW	15	95
rf electric power, MW	150	150
Other auxiliary power, MW	75	75
Net electric power, MW	1200	985
Cost of primary coolant loop, \$M	45	102
\$/kWe (relative)	1.0	1.2

expensive with helium than with water. The net effect is that the cost per unit power is ~ 20% higher with helium than with water for the typical conditions in STARFIRE. Several other areas of comparison between the two coolants were considered but they were found to be less important than the areas discussed above.

This study concludes that for the reference STARFIRE conditions the cost per unit power is ~ 20% higher with helium than with pressurized water cooling. Helium has many attractive features but it is not economically competitive if constraints derived from present knowledge are imposed. Several requirements for effective helium utilization can be identified. The most important of these are the development of a high temperature (> 600°C) structural material compatible with practical levels of impurities in helium and resolving the problem of void space in the inner blanket.

Heavy water (D_2O) has several advantages compared to H_2O. Processing of tritium from the water coolant is less difficult for D_2O and deuterium leakage from the first-wall coolant into the plasma chamber is less detrimental than hydrogen. Another important advantage of D_2O relates to the lithium burnup and energy distribution in the solid breeder. As discussed in more detail in later sections, the D_2O gives a more uniform burnup and energy distribution. The major disadvantage of D_2O is its high cost. This led to the selection of H_2O.

3.1.3 Materials Selection

The development of the reference STARFIRE first-wall/blanket design involved numerous tradeoffs in the materials selection process for the breeding material, coolant, structure, neutron multiplier, and reflector. With the limited scope of the present paper, only the major parameters and properties that impact materials selection and design criteria are discussed. Additional details are given in the STARFIRE design report.[1] Table 3 summarizes the primary candidate materials considered for STARFIRE and indicates the materials selected for the reference design.

Table 3. Candidate and Reference First-Wall/Blanket Materials

Breeder	Coolant	Structure	Neutron multiplier	Reflector
A. α-$LiAlO_2$	Pressurized water	Austenitic SS (adv. alloy)	Zr_5Pb_3	Carbon
B. γ-$LiAlO_2$	Pressurized water	Ferritic steel	Be	D_2O/SS
Li_2TiO_3			Zr	H_2O/SS
Li_2SiO_3			BeO	ZrC
			Pb-Bi eut.	
C. Li_7Pb_2	Helium	Ti alloy	PbO	
Li_2O		V alloy	Pb	
Li_2ZrO_3		Ni alloy	Bi	

A. Reference material for STARFIRE.

B. Other primary candidate materials.

C. Materials assessed but not candidates for STARFIRE.

Tritium-Breeding Material

The STARFIRE study has focused on the use of solid tritium breeding materials; and hence, liquid lithium, liquid lithium alloys, and molten salts have not been considered. 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. It is one of the most stable compounds considered and compatibility should not be a major problem. Adequate breeding is attainable with the aid of a neutron multiplier and the tritium release characteristics are nearly as good as any of the candidate compounds. The primary advantages of α-$LiAlO_2$ compared to γ-$LiAlO_2$ relate to the higher density, which will result in a thinner breeding zone, and the fact that α is the stable phase at temperatures below ~ 900°C. The major disadvantage of Li_2TiO_3 is a lack of data base. A slight potential advantage of this compound is its lower long-term activation compared to the aluminate. The silicate is similar to the above compounds, but because of its lower melting temperature, its chemical stability and compatibility characteristics are not as good.

Table 4 summarizes the allowable operating temperature ranges for the candidate compounds that 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 (~ 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 indicated.

Table 4. Predicted Temperature Limits for Adequate Tritium Release from Solid Breeding Materials

Material	Melting Temp., °C	Unirradiated T_{min}, °C	Unirradiated T_{max}, °C	Irradiated T_{min}, °C	Irradiated T_{max}, °C
Li_2O	1700	360	1000	410	910
$LiAlO_2$	1610	450	1000	500	850
Li_2SiO_3	1200	370	900	420	610[a]
Li_2Si	760	430	550	480	420
LiAl	700	250	500	300	380
Li_7Pb_2	726	270	530	320	390

[a] 510°C if significant burnup of lithium occurs.

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 than 100 wppm. Since this concentration translates to more than 35 kg of tritium in the blanket, Li_2O was not selected for STARFIRE.

Selection of the Structural Material

Six classes of materials generally considered as candidates for the first-wall/blanket structure are austenitic stainless steel, high-nickel alloys, titanium alloys, vanadium alloys, niobium alloys, and ferritic steels. Although the structural materials limitations were an important consideration in the selection of other blanket materials, the justification given here for the structural material choice is based on the specified coolant and breeder material. Nickel alloys are eliminated primarily on the basis of poor radiation damage resistance (embrittlement), no physical property advantage (thermal stress factor), and limited mechanical property advantage at temperatures required for water coolant. Titanium alloys are not considered viable candidates for the first-wall region because of their affinity for hydrogen. Vanadium and niobium alloys were eliminated because of their poor corrosion resistance in water and limited mechanical advantage at low operating temperatures.

The major focus has been on the tradeoffs between austenitic stainless steels and the ferritic steels. Table 5 summarizes the comparative advantage of an advanced austenitic stainless steel and a ferritic steel for the structural material. In addition to the design specific considerations listed in Table 5, the major drivers in the selection of the austenitic stainless steel relate to its ease of welding, its nonmagnetic properties, and the potential increase in the DBTT of ferritic steel after irradiation.

Neutron Multiplier

In order to achieve adequate tritium breeding with the primary candidate breeding materials, a neutron multiplier must be incorporated in the blanket. The most promising neutron multiplier materials are listed in Table 3.

Table 5. Comparative Analysis of Advanced Austenitic Stainless Steel versus Ferritic Steel

Advantages of austenitic stainless steel:

- Ease of welding
  - no post-weld heat treatment
  - higher reliability
- Mechanical properties less sensitive to composition and heat treatment
- Lower DBTT
- Lower corrosion mass transfer in water
- Non-magnetic

Advantages of ferritic steel:

- Lower radiation swelling
- Lower radiation creep
- Better physical properties — thermal conductivity and thermal expansion
- Moderately lower activation

Design specific considerations:

- Steady-state operation tends to reduce physical property (thermal stress factor) advantage of ferritic steel
- Water coolant and lower structure temperature tends to reduce radiation damage advantage of ferritic steel
- Long-term activation (> 30 yr) results primarily from molybdenum (~ 1% in ferritic steel; 2% in austenitic steel)
- Burnup of solid breeder poses additional blanket lifetime limitation

The lead and bismuth are acceptable in most respects, but the lead-bismuth eutectic alloy is preferred to either lead or bismuth. The lead-oxide is less desirable because of its very low thermal conductivity, high mass (relatively large thickness required), and compatibility problems. The beryllium oxide is considered the optimum multiplier for concepts in which the multiplier is homogeneously mixed with the breeder. The primary concern with zirconium is the marginal multiplication. That for lead-bismuch is the fact that it is a liquid liquid metal which has some compatibility problems at high temperatures. Beryllium provides high multiplication but is subject to radiation swelling and relatively high burnup, produces significant gas from transmutations, and produces relatively high burnup gradients in the blanket. A major advantage of beryllium is its light weight and relatively thin multiplier zone requirement. An attempt to find a material with the benefits of lead but which is solid at anticipated operating temperatures led to the consideration of the compound Zr_5Pb_3. Although the data base is limited, this material reportedly melts at 1400°C and has a calculated density of 8.9 g/cm^3. Beryllium is identified as a backup for the neutron multiplier.

Reflector

Primary candidates for the reflector are H_2O or D_2O contained in austenitic steel, graphite and carbides such as SiC or ZrC. The H_2O/stainless steel provides the thinnest reflector region. Graphite requires a thicker reflector zone but results in minimal activation products. With both the water and graphite, an austenitic steel with low molybdenum content is proposed to reduce long-term activation. Stable carbides are considered to mitigate the potential for carburization of the steel at high temperatures.

3.1.4 Design of Blanket Modules

Several design concepts have been considered for integrating the breeder, structure, coolant, neutron multiplier, and reflector into an optimized blanket design. The addition of each of these components significantly impacts the design and material compatibility problems that must be considered. Three blanket concepts were analyzed in the selection of the reference design for STARFIRE. The first concept utilizes a separate neutron multiplier zone between the first wall and the breeder region. The second concept utilizes a heterogeneous arrangement of several neutron multiplier and breeder regions behind the first wall followed by a separate breeder region. The third concept utilizes a homogeneous neutron multiplier/breeder region directly behind the first wall followed by a separate breeder region. This last concept requires compatible breeder/multiplier materials, which probably limits the multiplier to beryllium oxide. Similar first-wall and reflector designs are proposed for all three concepts. The second concept is more complex from a design point of view and probably requires a larger structural fraction to separate the breeder regions from the multiplier regions. The first concept with the separate multiplier region is selected for the reference design, primarily because a more uniform lithium burnup and energy generation rate can be obtained in the blanket. This advantage allows longer blanket life before changeout and simplifies the heat transfer problems.

Figure 2 is a schematic diagram of the blanket cross section showing the water-cooled austenitic stainless steel first wall, the Zr_5Pb_3 neutron multiplier zone separated from the breeder region by a water-cooled panel, the water-cooled $LiAlO_2$ breeder region, and a graphite reflector. First-wall blanket design parameters are summarized in Table 6.

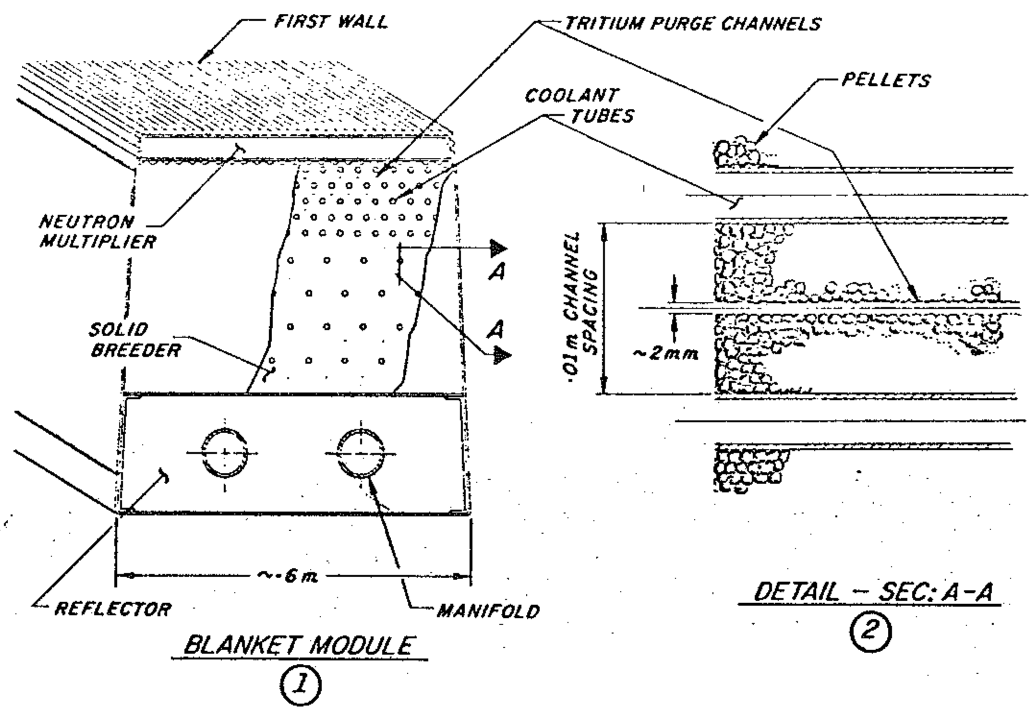

Fig. 2. Schematic diagram of STARFIRE blanket concept.

Table 6. Summary of Blanket Design Parameters

First-wall/blanket parameters	
Structural material	Advanced austenitic stainless steel
Structural wall thickness, mm	1.5
Maximum structural temperature, °C	< 450
Coating/cladding	Beryllium
Coating/cladding thickness, mm	1.0
Coolant	Pressurized water
Coolant outlet temperature, °C	320
Coolant inlet temperature, °C	280
Coolant velocity, m/s	10
Breeding region	
Structural material	Advanced austenitic stainless steel
Maximum structural temperature, °C	425
Breeder material	α-$LiAlO_2$
Theoretical density, g/cm^3	3.4
Effective density, %	60
Grain size, 10^{-6} m	0.1
Maximum temperature, °C	850
Region thickness, m	0.4
Coolant	Pressurized water
Coolant inlet temperature, °C	280
Coolant outlet temperature, °C	320
Neutron multiplier	
Material	Zr_5Pb_3 (or Be)
Maximum temperature, °C	900
Thickness, m	0.07
Density, g/cm^3	8.9
Reflector	
Material	Graphite
Thickness, m	0.15
Maximum temperature, °C	< 900
Structure	Low molybdenum stainless steel
Structure temperature	< 450°C

First Wall and Blanket Structure

The water-cooled austenitic stainless steel first wall is a panel coil-type construction and is an integral part of the blanket module. The coolant temperature is maintained between 280 and 320°C throughout the first wall and blanket. For the average neutron wall loading of 3.6 MW/m^2, the average surface heat flux on the first wall is 0.92 MW/m^2 with a peak-to-average value of $\sim$ 1.2. The maximum structural temperature in a 1.5-mm thick stainless steel wall is 450°C for the reference conditions. For these low temperatures, an estimated wall design life of $\sim$ 16 MW/m^2 is considered reasonable for an advanced austenitic stainless steel.

The proposed panel coil-type construction provides integral cooling of the blanket wall and avoids the necessity for a large number of tube welds in the high radiation zone. Since fabrication by a roll-bonding process does not greatly affect the microstructure of the steel, radiation damage in the weld region should not differ substantially from the bulk material. Also, the panel coil structure is perceived to have less vibration problems than an unsupported tube bank.

A $\sim$ 1-mm thick beryllium coating or clad on the first wall serves to protect the plasma from the high-Z wall material. This thickness will provide sufficient material to withstand the surface erosion for the blanket lifetime of $\sim$ 6 yr. A slightly thicker beryllium coating may be required on the inboard wall to accommodate the projected number ($\sim$ 10 per wall lifetime) of plasma disruptions.

Tritium Breeding Zone

The $\sim$ 40-cm tritium-breeding zone consists of a packed bed of α-$LiAlO_2$ with 1-cm diameter stainless steel coolant tubes spaced appropriately throughout the zone to maintain a maximum breeder temperature of 850°C. The tube spacing increases from $\sim$ 2 cm at the front of the breeder zone to 5-10 cm at the back. The coolant inlet temperature is 280°C with an outlet temperature of 320°C. The relatively low temperature of the austenitic stainless steel tubes (< 400°C) and the oxide film on the water side of the tubes provide an adequate tritium barrier for inleakage into the coolant. The $LiAlO_2$ is perforated with $\sim$ 2 mm diameter holes through which low-pressure pressure ($\sim$ 1 atm) helium passes to recover the tritium from the breeder. The $LiAlO_2$ is $\sim$ 60% dense to facilitate percolation of tritium (as T_2O) to the helium purge channels. Tritium generated within $LiAlO_2$ grains diffuses to the surface of the grains, desorbs from the grain surface as T_2O, migrates along interconnected grain boundary porosity to the surfaces of the breeder particles, and finally percolates through the particle bed to the helium purge channels where it is carried to the processor.

Maximum lithium burnup in the blanket region will be $\sim$ 25% for a first-wall lifetime of 20 MW-yr/m^2. In addition to changing the stoichiometry of the breeder material, breeding characteristics and the energy generation profile in the blanket will be affected. When a neutron multiplier is used, the tritium breeding comes almost exclusively from ^{6}Li. Therefore, highly enriched lithium (> 50% ^{6}Li) is used to minimize the lithium and tritium inventories. Although some tradeoffs are possible, a limit of $\sim$ 20 MW-yr/m^2 is reasonable for the breeder. This restriction tends to limit the value of a longer lifetime structure for a solid breeder blanket.

Neutron Multiplier

The proposed Zr_5Pb_3 neutron multiplier provides some of the benefits of a lead multiplier while maintaining the design simplicity of the solid state materials. These advantages include a more uniform burnup and energy generation rate in the blanket region. The maximum lithium burnup in the blanket region is ~ 50% higher with a beryllium multiplier. Burnup of the multiplier is also reduced with the heavier elements (zirconium and lead) since the n,2n reaction simply leads to the formation of another isotope of the same element in most cases. A multiplier zone thickness of ~ 50% higher with a beryllium multiplier. Burnup of the multiplier is also reduced with the heavier elements (zirconium and lead) since the n,2n reaction simply leads to the formation of another isotope of the same element in most cases. A multiplier zone thickness of ~ 7 cm is required to provide sufficient multiplication. The back side of the first wall and water-cooled panel between the multiplier and breeder zones provides cooling for the 7-cm slab. Approximately 30% of the neutron heating is deposited in the multiplier zone with maximum temperatures of the Zr_5Pb_3 calculated to be ~ 900°C. Additional blanket structure will be required to support the ~ 5×10^5 kg (500 tons) of neutron multiplier.

Reflector, Manifold, and Structural Support

The reflector consists primarily of ~ 15 cm of graphite. The 20-cm support structure to which the blanket modules are attached also serves as the containment for the graphite reflector. In order to conserve space, the manifolding for the blanket is imbedded in the graphite reflector. The manifolding with appropriate additional channels serves as the coolant for the reflector region. A modified austenitic steel with low molybdenum content is used in this low-flux region to reduce the long-term activation.

Total Module

The total blanket module, with a thickness of 68 cm, consists of a 1-cm thick first wall, a 7-cm thick neutron multiplier, a 40-cm thick breeding zone, and a 20-cm thick reflector that contains the blanket support structure and the manifolding. The modules are 2-3 m wide by ~ 3 m high depending on the location within the reactor. The module walls and all support structures in the high-radiation zone are fabricated from an advanced low-swelling austenitic stainless steel. All internal structure is integrally cooled to remove the nuclear heating and maintain the structure below 400°C.

For plasma stability, an electrical conducting path equivalent to 2 cm of stainless steel is required near the first wall. The conductivity of the first wall and the neutron multiplier meets this requirement in the modules. Between modules, a conducting path in the wall of the module to the back of the blanket and across a jumper to the next module is provided to complete the electrical circuit.

3.2 Limiter Design

The limiter/vacuum system uses a toroidal limiter, 1 m high, centered at the midplane and located at the outer periphery of the plasma (see Fig. 3). The toroidal limiter consists of 96 sectors. Each sector has a mushroom-shaped cross section that consists of symmetrical top and bottom flat ribbon sections joined to the coolant inlet and outlet headers.

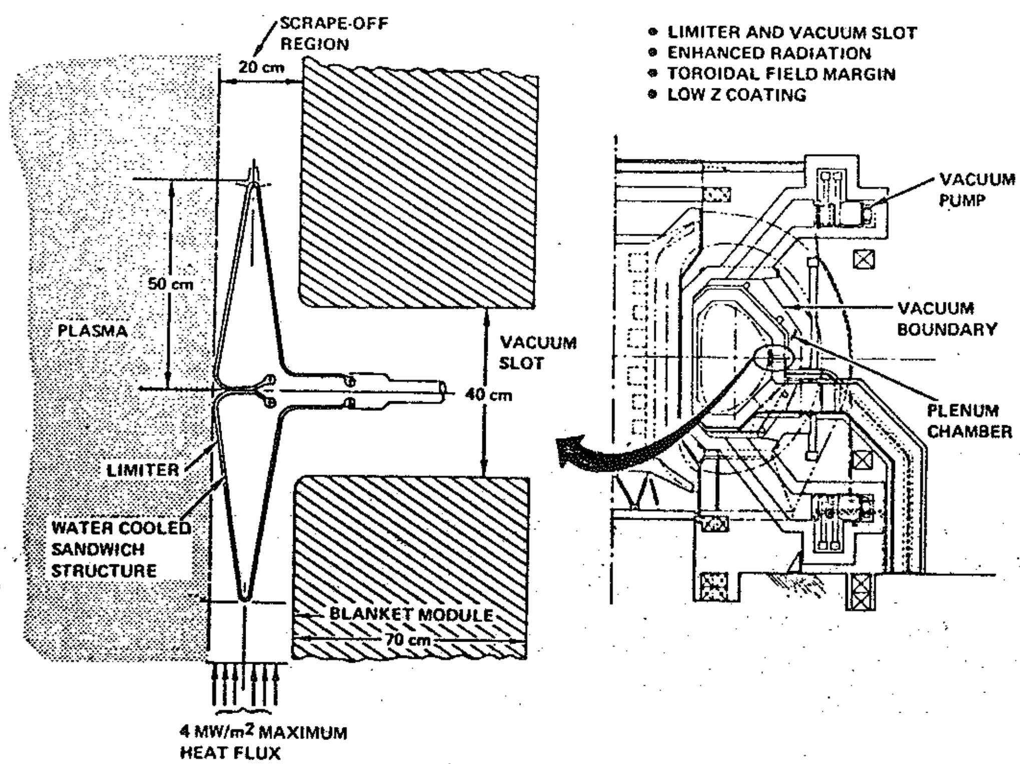

Fig. 3. Schematic of the impurity control and exhaust system showing the limiter design.

The heat load on the limiter consists of three components: (1) transport power flux given by $q = 16 \sin\theta \, e^{-x/\delta_E}$ MW/m^2, where θ is the angle that the poloidal field line makes with the limiter and $\delta_E = 5$ cm; (2) heat flux from plasma radiation and neutrals ~ 0.9 MW/m^2; and (3) volumetric nuclear heating of ~ 40 MW/m^3. A maximum surface heat load of 4 MW/m^2 occurs at the two leading edges (one at the top and one at the bottom) where $\theta = 0$. Moving the leading edge closer to the first wall reduces the peak heat load but also reduces the number of alpha and impurity particles that can be pumped. The location of the leading edge at $x = 7$ cm was determined from a tradeoff analysis.

A detailed assessment of material candidates that included radiation effects, thermal-hydraulics, and stress analysis was performed. The relatively high surface heat load and the intense radiation environment require a structural material with good thermomechanical properties and resistance to radiation damage. The use of liquid metal coolants permits several excellent structural materials, particularly the refractory alloys, to be considered. Since liquid metals were excluded because of perceived safety problems, only water offers an attractive limiter coolant. The choice of water as a coolant limits the structural material options somewhat. A tantalum alloy is selected as the primary structural material with vanadium and copper alloys as backup options. Both tantalum and vanadium permit useful heat recovery for energy conversion, while with copper the permissible coolant pressure and temperature are so low that useful heat recovery is not feasible.

The water coolant speed is taken as ~ 8 m/s to assure a heat transfer coefficient of $\sim 10^4$ Btu/hr-ft^2 °R. In addition, liquid subcooling of 55°C is assumed to prevent transition from subcooled nucleate nucleate boiling to film boiling at the leading edge. The coolant temperature rise and pressure drop per pass are 15°C and 0.2 MPa, respectively. The design is based on a double pass (two sectors) with coolant inlet temperature of 135°C into one sector and a coolant exit temperature from the adjacent sector of 165°C. The maximum (inlet) coolant pressure is 2.8 MPa. The 200-MW heat from the limiter is used for feedwater heating in the power conversion cycle. The critical area of the limiter from a thermal stress standpoint is the leading edge which receives the highest surface heat flux. The design of the leading edge must ensure that stresses are low enough to preclude failure over the lifetime of the limiter. Detailed stress analysis shows that the maximum effective stress due to combined temperature effects and coolant pressure is ~ 315 MPa at the leading edge for a limiter outer wall thickness of ~ 1-1.5 mm. This is acceptable for the tantalum alloy. Table 7 shows the major features of the limiter design.

Table 7. Major Features of the Limiter Design

Structural material	tantalum alloy
Low-Z coating material	beryllium
Coolant	water
Coolant inlet temperature, °C	135
Coolant outlet temperature (2 pass), °C	165
Maximum coolant pressure, MPa (Psia)	2.8 (400)
Total heat removed from limiter, MW (90 MW transport, 56 MW radiation plus neutrals and 54 MW nuclear)	200
Maximum heat load (at leading edge), MW/m^2	4
Coolant channel size	8 mm x 4 mm
Wall thickness, mm	1.5
Ratio of maximum effective stress to the allowable	0.9
Maximum material temperature (coolant side), °C	235
Maximum material temperature (coating side), °C	350
Maximum nuclear heating rate, MW/m^3	79
Atomic displacements, dpa/yr	14
Helium production rate, appm/yr	< 5
Hydrogen production rate, appm/yr	37

4.0 Power Plant Considerations

4.1 Heat Transport System

The thermal energy deposited in the blanket, first wall and limiter is converted to electricity in the steam power conversion system. The primary function of the heat transport system is to remove this thermal energy from the reactor and deliver it to the power conversion system while maintaining the temperature of the first wall, blanket and limiter within specified limits. In addition, this system has the capability to remove the low level of residual heat generation in the blanket when the reactor is shut down.

Two separate heat removal systems operating at different pressure and temperature levels are utilized. The blanket/first wall primary loops cool the breeding blanket and the first wall, an operation simplified by the integral design of these reactor components. The power deposited in the limiter, approximately 5 percent of the total thermal power, is used for feedwater heating in the steam power conversion system and is removed by the limiter/feedwater loop. Systems studies have indicated that recovery and conversion of this limiter thermal energy, even though it is at a lower temperature than the primary loop, is economically attractive. Fusion thermal energy removed from other reactor components (e.g., shield, rf waveguides, structure) is of such small quantities or at such low temperature that its utilization in the power conversion system is not feasible. For the particular materials and temperature constraints adopted for STARFIRE, water is the preferred coolant and is utilized in both the blanket/first wall and limiter cooling circuits.

4.1.1 Blanket/First-Wall Primary Loops

The blanket/first wall dual primary loops consist of piping and valves, pumps, pressurizers, steam generators, water conditioning equipment and instrumentation and controls. The system is shown schematically in Fig. 4 and major parameters are given in Table 8. Because STARFIRE operates steady-state rather than in a pulsed mode, a thermal energy storage system is not required. Moreover, it is believed feasible to keep the tritium concentration in the primary coolant low enough such that an intermediate loop is not needed.

Primary coolant leaves the blanket at 320°C and is returned at 280°C. This 40°C ΔT, which is comparable to that in pressurized water fission reactors, was selected to be compatible both with a desirable feedwater temperature into the steam generator and with blanket thermal design considerations. The pressure throughout the primary loop is maintained in the 14.6-15.2 MPa (2100-2200 psi) range.

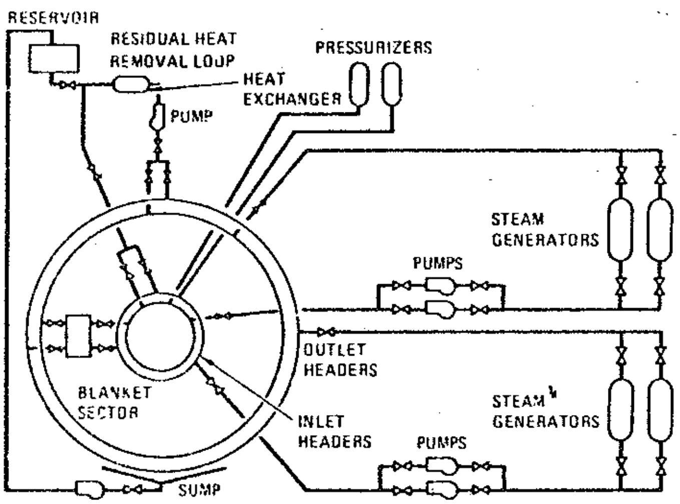

Fig. 4. Schematic of primary loops with residual heat removal loop.

Dual lines are utilized to connect the primary loop components. The piping, which is constructed of stainless steel or carbon steel with stainless steel internal cladding, is sized for a maximum water velocity of 20 m/s. The distance between the reactor and the steam generators located inside the reactor building is about 25 m, and the pressure drop in this pipe run (and indeed throughout the system) is relatively small.

Four steam generators rated at about 950 MW each are utilized to provide 7.3 x 10^6 kg/hr of slightly superheated steam at 299°C and 6.3 MPa (800°F, 910 psi) to the turbines. These units are conventional, vertical, tube-and-shell units and are designed with state-of-the-art PWR technology, incorporating those modifications necessary to make them compatible with a "total remote maintenance" philosophy. The tube material is Inconel. The steam generators are approximately 24 m long by 2.7 m diameter with a mass of about 300,000 kg each.

Table 8. Heat Transport System Parameters

Blanket/First Wall Primary Loops	
Coolant	Water
Heat Load	3800 MW
Pressure	15.2 MPa (2200 psi)
No. of Independent Primary Loops	Two
Pipe Size	0.93 m ID
Maximum Velocity	20 m/s
Pumping Power	20 MW
Coolant Volume	∿ 500 m^3
No. of Steam Generators	Four
No. of Pumps	Two per loop
Limiter Feedwater Loop	
Coolant	Water
Heat Load	200 MW
Pressure	2.8 MPa (400 psi)
Maximum Coolant Temperature	165°C
No. of Loops	One
No. of Feedwater Heaters	Three
No. of Pumps	Two

The electric motor driven primary pumps circulate the water coolant at a rate of approximately 60 x 10^6 kg/hr, developing a head of about 60 m. Two pumps are provided as shown in the schematic, each with 50 percent capacity. In the event one pump must be removed from service for maintenance, this can be accomplished while the reactor operates at reduced power with the other unit.

Appropriate pressure levels are maintained in the system through the use of electrical heaters and condensing water sprays in the pressurizer. A water conditioning system maintains the proper water purity and chemical concentrations, and is the system from which a small bleed flow of water is directed to the tritium facility for primary loop tritium control. Valving is provided so that an entire primary loop or its individual components can be isolated as desired for inspection and servicing. With this arrangement it is possible, for example, to isolate an individual steam generator (if it develops a leak, for instance) and to run the reactor at reduced power while repairs are made.

4.1.2 Limiter Feedwater Loop

Thermal energy deposited in the limiter is transported via the limiter feedwater loop to feedwater heaters. The limiter feedwater loop shown schematically in Fig. 5 incorporates piping and valves, pumps, pressurizer, the feedwater heaters, a water conditioning system and instrumentation. As indicated in Table 8, a single water loop operating at a nominal pressure of 2.8 MPa (400 psi), with limiter inlet and outlet temperatures of 135°C and 165°C,

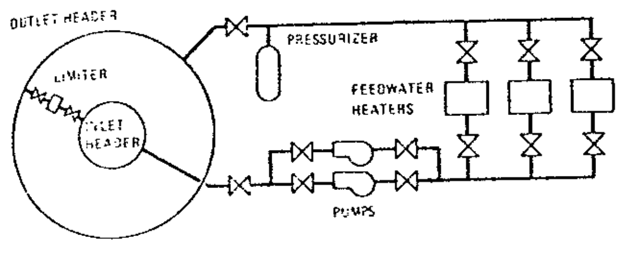

Fig. 5. Limiter feedwater loop schematic.

respectively, removes 200 MW of thermal power from the limiter. Piping approximately 30 cm in diameter is utilized to limit the maximum velocity to about 20 m/s.

The steam power conversion system described in Sec. 4.2 incorporates three feedwater trains, each with eight feedwater heaters. Each train, in addition to the conventional heaters, a unit which utilizes limiter coolant as the heat source.

4.1.3 Residual Heat Removal Loop

When the reactor is shut down, the afterheat generation rate in the blanket is large enough to require active cooling. It is emphasized, however, that while detailed calculations have not been done, preliminary analysis indicates that the need for cooling the blanket in the shutdown mode is not a safety consideration, but rather one of preventing blanket damage to protect the capital investment. At shutdown, the maximum local rate of temperature rise without cooling is about 0.15°C/s, yielding 90°C rise in 10 minutes. Because the afterheat decays rapidly, this temperature rise rate is reduced significantly in a very short time.

Immediately after shutdown, the residual heat removal function is performed by one of the primary loops by keeping one steam generator and one pump operating. However, when the heat load becomes quite small, continued operation of a steam generator may no longer be feasible, and this function would be performed by the residual heat removal loop.

This loop would normally operate with the primary loops isolated. As shown in Fig. 4, pumps, valves, a heat exchanger and a reservoir are incorporated into this system. It is noted that through proper valving, the flow may be directed through either one or both of the blanket circuits. Moreover, sumps are provided such that in the unlikely event a leak develops in both primary loops it is possible to maintain cooling water flow to the blanket.

4.2.1 System Functions

The power conversion system, shown schematically in Fig. 6, it utilized to convert the reactor thermal energy to electrical power. The 4000 MW of thermal energy generated by the reactor is transferred to the power conversion system by the Heat Transport System, as described in Sec. 4.1. The principal functions of the system are:

1. To convert the available thermal energy to electrical energy, at the highest thermodynamic efficiency economically possible.
2. To reject to the atmosphere the non-recoverable thermal energy.

Principal parameters of the power conversion system are presented in Table 9.

4.2.2 General Description

The power conversion system consists of components of conventional design for use in large central generating stations. The thermodynamic cycle itself and its components resemble the thermal cycle of a pressurized water reactor plant.

Turbine plant equipment is selected for high operating efficiency, to provide maximum generating capability and reliability. All equipment is designed for continuous operation at 105 percent of the rated steam flow. The subsystems and/or components that comprise the power conversion system are:

1. The main steam supply system,
2. The turbine-generator unit,
3. The moisture separator/reheaters,
4. The heat rejection system,
5. The condensate system,
6. The feedwater system.

4.2.3 Subsystem/Component Description

Main Steam Supply System

The main steam supply system transports the superheated steam from the four steam generators to the high pressure stage of the main steam turbine's throttle/stop valves. Piping will interconnect the steam generators and supply the steam with the minimum economical pressure losses, assuring uniform heat removal from each steam generator, thus maintaining system balance. In addition, this piping arrangement provides mixing to ensure uniformity of steam condition at the inlet to the high pressure turbine. Branches from the main steam system are used to provide heat to the moisture separator/reheater.

The main steam lines are designed with an isolation valve and a check valve in each line outside the reactor building. The isolation valves will close automatically on high steam flowrate simultaneously with low temperature or low line pressure, or on a high reactor building ambient pressure signal, and can also be operated remote-manual from the control room or local panel. The check valves prevent backflow of steam into the steam generator

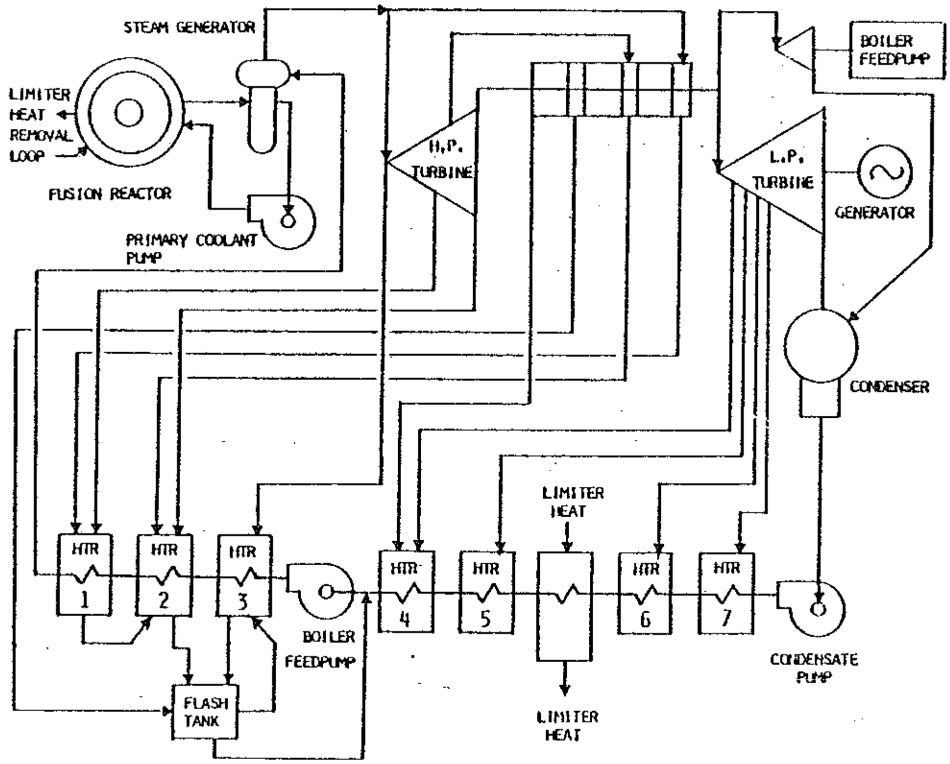

Fig. 6. Power conversion system.

Table 9. Power Conversion System Parameters

Thermal Power	4000 MWth
Net Electrical Power	1440 MWe
Gross Turbine Cycle Efficiency	36%
Steam Flow Rate	7.25×10^6 kg/hr
Steam Temperature	299°C
Steam Pressure	6.3 MPa

and the reactor building in the event that a loop is not operating for either maintenance or emergency reasons.

Turbine-Generator Unit

Turbine - The turbine consists of one double flow high pressure stage in tandem with three double-flow low pressure stages. The turbine is a four casing 1800 rpm unit with tandem-compound six flow exhaust.

The unit is provided with two steam chest assemblies, one located on each side of the high pressure turbine. Each assembly consists of two throttle stop valves and two governing valves. Each valve is controlled by an electrohydraulic governing system through an individually operated valve actuator.

Generator - This equipment includes a hydrogen inner cooled synchronous generator with a water cooled stator and a shaft driven, air cooled brushless exciter. The unit has a power factor of 90 percent with output of 22,000 to 25,000 volts, three-phase, 60 Hz.

Moisture Separator/Reheater

The moisture separator/reheater removes the moisture of the wet steam exhausted by the high pressure turbine and reheats the steam, providing approximately 40°C (100°F) of superheat. The wet steam at about 8 percent moisture content, enters the moisture removal section where 100 percent of the moisture is removed and drained to the feedwater system for heating. The dried steam then passes through the two element reheater section, where it is first heated by extraction steam from the high pressure turbine and second, heated by main steam. The condensed steam and the moisture removed are used in the feedwater system for heating. The superheated steam from the reheater enters the low pressure turbine and the feedwater pump turbine.

Heat Rejection System

A multi-pressure three stage condenser with a 5.6°C (10°F) rise for each stage is provided, for a total rise of 16.7°C (30°F). The average backpressure on the condensing steam will be 6.75 kPa (2" HG). Each condenser stage will be located below one of the low pressure turbine stages, taking the exhausted steam directly from that particular turbine stage.

Three circular cooling towers have been selected to minimize space requirements. The towers will be either mechanical draft or hyperbolic natural draft or a combination of both. The tower configuration will be determined by economics and reliability. Approximately 63,100 l/s (1,000,000 gpm) will be circulated through the cooling towers and condenser.

Condensate System

The condensate pumps take suction from the condenser hot well, transferring the condensate to the feedwater system. The main function of the system is to condition the condensate by deaeration, polishing, chemical addition and condensate make-up to provide feedwater at the purity required for the steam generators and turbine optimum performance and reliability.

Feedwater System

The feedwater system supplies feedwater to the steam generators at the required flow rate, temperature and pressure for optimum turbine output and thus optimum cycle efficiency. The system will consist of three trains of heaters and pumps, each train designed for 35 percent of rated flow. There are a total of eight sets of heaters, five low pressure heaters and three high pressure. One set of low pressure heaters is heated by the 200 MW available from the limiter.

The three normally operating feedwater pumps are turbine driven. A fourth motor driven pump, also designed at 35 percent of rated flow, is provided as a spare and for startup.

An auxiliary feedwater system can be provided, if required for capital investment protection.

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