Blanket Concepts for the ARIES Commercial Tokamak Reactor Study

S. P. Grotz
University of California, Los Angeles, CA USA
T. Hua, D. K. Sze
Argonne National Laboratory, Argonne, IL USA
E. T. Cheng, R. L. Creedon, C. P. C. Wong
General Atomic Company, San Diego, CA USA
J. S. Herring
Idaho National Engineering Laboratory, Idaho Falls ID USA
A. Klein
Oregon State University, Corvallis, OR USA
L. Snead, D. Steiner
Rensselaer Polytechnic Institute, Troy, NY USA
N. M. Ghoniem, M. Z. Hasan, R. C. Martin, F. Najmabadi, S. Sharafat
University of California, Los Angeles, CA USA

INTRODUCTION

The ARIES study (Conn and Najmabadi, 1987 and Najmabadi, 1988) is a 3-year effort, started in 1988, exploring the potential of the tokamak to be an attractive and competitive commercial power reactor. Several different versions of the tokamak are being considered, combining different levels of extrapolations in physics and engineering databases. The first version studied in detail, ARIES-I, combines present-day physics (with minimal extrapolation) with aggressive engineering technology such as very high-field, superconducting magnets and low-activation silicon carbide composite materials. The ARIES-I version is designed to meet acceptable safety and environmental criteria. In particular, achieving a passively safe concept that meets Class-C waste disposal is one of the high leverage items in the design. This paper summarizes the scoping analysis and engineering design of the ARIES-I fusion-power-core subsystems. The ARIES-I design is a $1000\,MW_e$ power reactor, operating at steady state in the 1^{st} stability regime and uses a high magnetic field. Typical operating parameters of the ARIES-I "strawman" design are listed in Table-I.

FIRST WALL AND BLANKET CONFIGURATION

The blanket chosen for ARIES-I is a gas-cooled ceramic design. The structural material for the first wall/blanket/shield (FW/B/S) is a silicon carbide (SiC) composite with woven SiC fibers in a matrix of SiC. The plasma-side of the first wall has an additional 2 millimeter coating of SiC which is applied by chemical vapor deposition. The coating serves as an erodible layer over the lifetime of the FW/B/S. The tritium breeding material is lithium oxide (Li_2O), chosen primarily because it offers the potential to have adequate tritium breeding without the use of a neutron multiplier. The Li_2O is in the form of plates with a SiC coating. This coating prevents erosion by the primary coolant.

The coolant for the FW/B/S is carbon dioxide (CO_2) which carries very fine particulates of SiC (or SiO_2 as an alternate). The mixture of CO_2 and particulates results in increased effective heat capacity of the primary coolant. The increase in heat capacity allows the system to operate at a reduced pressure of 0.5 MPa which permits the use of large blanket modules for a given allowable design stress in the SiC-composite. The coolant inlet and outlet temperatures are 250 °C and 700 °C, respectively. After removing heat from the FW/B/S, the coolant passes through the heat exchanger and then a series of cyclone separators to remove the particles from the CO_2 prior to entering the primary coolant circulator/blower. After the CO_2 leaves the circulator/blower, the particulates are mixed with the gas and the mixture is sent back to the FW/B/S.

A poloidal cross section of the ARIES-I fusion power core is shown in Figure 1. Typical operating parameters are given in Table 1.

MATERIALS

One of the primary goals of the ARIES-I scoping phase was to select materials that meet the Class-C waste disposal conditions (shallow land burial) and use these materials in such a way that the reactor would be considered passively safe. It has been shown that SiC is a low-activation material (Hopkins, et al., 1983) that meets the Class-C regulations, provided that the level of impurities (primarily iron) is well controlled. Also, the level of afterheat at shut down is extremely low, hence passive safety appears to be credible. Recent advances in composite materials development have produced a variety of composite materials in which fibers are imbedded in a matrix of similar and dissimilar materials. The specific strength of these composites is often greater than the matrix material alone. Because of the brittle nature of monolithic SiC, the tensile strength is often the limiting factor in blanket design. A composite material with SiC fibers in a SiC matrix has been shown to be much stronger than an equivalent thickness of monolithic SiC. Failure strengths as high as 750 MPa have been measured (Fitzer, et al. 1986) in SiC-composites at elevated temperatures of 1000 °C to 1400 °C, whereas monolithic SiC has a failure strength of about 350 MPa at 1000 °C.

Physical erosion of the FW/B/S components is a concern when the CO_2/particulate system is employed as the primary coolant. Several particulate materials are being considered. To avoid a "sand-blasting" effect on the structures in contact with the coolant the following guidelines were chosen:

1. Use materials for the particulates which are softer than the materials in contact with the primary coolant;
2. Use small-sized particles, 5 to 10 μm diameter;
3. Orient the flow channels so that flow is largely parallel to the wall and, where bends are needed, provide large-radius bends;
4. Control the velocity of the coolant so that it is well below the threshold velocity at which severe erosion occurs, typically 100 m/s.

The primary candidates for the particulate material are either SiC or SiO_2.

The Li_2O breeding material is subject to excessively high rates of erosion, therefore a coating of SiC is applied to the outer surfaces of each breeder plate. The thickness of the coating will also provide additional strength to maintain the integrity of the breeder plate under irradiation induced swelling of the Li_2O. More details of the materials issues of ARIES-I are being studied (Sharafat, 1989).

NEUTRONICS

Neutronics investigations were performed on the selected CO_2-cooled, SiC ceramic-composite material blanket for the ARIES-I design. The nuclear performance parameters studied include: tritium breeding, blanket energy multiplication, radiation protection of superconducting magnets, and activation issues. Two poloidal and toroidal models (one dimensional) were used in the calculation. The poloidal-geometry model uses the plasma centerline as the axis of an infinite cylinder. The inboard and outboard blanket and shield composition will be identical in this model. The toroidal-geometry model uses the centerline of the fusion reactor as the the axis of an infinite cylinder. Based on this geometry, the inboard and outboard blanket and shield compositions can be modelled separately. The toroidal model is especially suitable for the ARIES-I design which has a high elongation of 2.24. The full-coverage tritium breeding ratio and blanket energy multiplication were calculated by using the toroidal model. The net performance of the design due to loss of surface coverage was estimated by the combination of these two models.

For the selected design, the first wall is 1.4 cm thick. The blanket is composed of 6.4 % SiC, 74.9 % Li_2O breeder and 18.7 % gas. A range of 0.5 to 2% by volume of solid particulate will be carried in the coolant. For this design, we found that at a blanket and shield thickness of 0.58 m and 0.8 m, respectively, the full coverage tritium breeding ratio is 1.19 and the blanket energy multiplication is 1.06. With no breeding behind the divertor the allowable surface coverage of the divertor is 14%, and the minimum inboard blanket and shield thickness to protect the superconducting magnet for a life-time of 60 years at a neutron wall loading of $5\,MW/m^2$ is $1.4\,m$. Natural enrichment of ^{6}Li can be used for the design and no neutron multiplier will be needed to obtain this performance. Negligible decay heat and very low shut down biological dose rate will be generated from this SiC ceramic-composite design.

THERMAL HYDRAULICS

The goals for the ARIES-I thermal-hydraulics design are to maintain the first wall maximum temperature below 1000 °C and the total first wall and blanket pumping power in the range of $2\,MW$. The coolant chosen is CO_2 (0.5 MPa) with a dilute suspension of particulates. When compared to a conventional gas-cooled design, the primary advantage of this design is to greatly improve the effective volumetric heat capacity of the coolant, allowing lower flow velocity, smaller pressure drops, and reduced coolant pressure while maintaining the same heat transfer capability. To achieve a uniform circulating flow, the particle and gas mixture is kept in the "dilute flow" regime. The velocity range of the coolant in the first wall and blanket is from 15 to $43\,m/s$. The average solid-to-gas mass ratio (weight of solid per unit weight of gas) is 14.

To maintain a high effective volumetric heat capacity, however, one would like to operate the coolant at a high solid-to-gas mass ratio. A range of values for the mass ratio has been studied. This mass ratio is limited to about 15 in order to prevent saltation or settling of the particles. These guidelines indicate the use of high effective gas density. Since for the same effective gas density, we can operate a higher density gas like CO_2 at a much lower pressure than a lower density gas like He, we have selected CO_2 as our carrier gas. Trade-off studies examining the sensitivity of the design to various operating parameters were performed (Wong and Hasan, 1989).

The primary thermal energy conversion scheme chosen for the ARIES-I reactor is a standard Rankine cycle with dual reheat. Cycle efficiency is 46 %. An alternate energy conversion scheme is under consideration (Hasan and Martin, 1989). This scheme employs dissociating gases as the working fluid. Gases under consideration are nitrosyl chloride ($NOCl$) and nitrogen tetroxide (N_2O_4). Efficiency of such a system could be 50 to 55% or higher, depending on the primary coolant inlet and outlet temperatures.

SAFETY

The extensive use of SiC throughout the fusion power core reduces the inventory of the activation products substantially. The tritium which is bred in the blanket and carried throughout the primary loop in the coolant is the primary source of off-site radiation in the event of a breach of all levels of containment. It is suggested that the SiC-coating on the breeder plate will act as one level of tritium containment and reduce the leakage rate of tritium into the primary coolant. This is particularly true if, at the onset of the breach of containment, the plasma is quenched and the blanket (with no afterheat) begins to cool. As the blanket cools, the diffusion coefficient for tritium in both the Li_2O and SiC are reduced and the tritium becomes less mobile.

The thermal response of the first wall to off-normal events such as loss-of-flow accidents (LOFA), loss-of-coolant accidents (LOCA), and disruptions is being investigated. The ceramic structure is not electrically conducting, so there should be no appreciable eddy currents induced in the FW/B/S during a disruption. The thermal analysis of the first wall during a disruption was studied using a one-dimensional finite element method. The decomposition temperature of SiC is about 2800 °C and at the end of the disruption an average of $20\,\mu m$ of SiC is lost over the entire first wall area. A peaking factor of 5 would result in local losses of about $100\,\mu m$. This analysis does not include the effects of vapor shielding, which is currently under study. The vapor shielding effect is expected to reduce the impact of the disruption heat loads. The LOFA and LOCA are currently under study to predict how long the plasma can remain on following the onset of the event. The most serious problem will be particulate settling which will greatly reduce the coolant heat capacity. It is expected that a long pump coast-down time at the onset of the LOFA will increase the time before settling occurs, thus allowing sufficient time to shut down the plasma.

The fusion power core meets the shallow land burial requirements for waste disposal if the radioactive inventory is volume averaged. Locally, near the first wall, ^{26}Al is produced and its concentration exceeds the requirements for shallow burial. The reaction which produces this isotope has a threshold at 13 MeV, thus the ^{26}Al concentration is only significant at the first wall, before the 14 MeV neutrons are attenuated.

TRITIUM SYSTEMS

The ARIES-I blanket operates in a condition very different from what is usually encountered in a fusion reactor. The combination of structural materials, coolant, and blanket tritium recovery method generate some unique problems for the blanket tritium systems. Some initial calculations have been made to assess critical issues. More detailed analysis will be performed to resolve these critical issues during the design phase.

The SiC structure has a very high hydrogen solubility and very low hydrogen diffusivity, particularly at low temperature. Therefore, at low temperature, the tritium implanted on the first wall will take a long time to "back diffuse" into the plasma. If the first wall temperature is below 600 °C, then it will take well over 100 *sec* until the rate of back diffusion equals the implantation rate. If the reactor starts up cold, then all of the tritium incident on the first wall will be absorbed, and a very large fueling system will be required. There are two ways to alleviate this problem: (1) Pre-heat the first wall and blanket and start the reactor hot, or (2) Coat the first wall with a high diffusivity, low-Z material.

The blanket tritium recovery is through the primary coolant. The tritium bred in the Li_2O will diffuse into the coolant and be recovered from a side stream. The elimination of a separate purge system greatly simplifies the blanket design. The addition of the particulates to the gas coolant greatly reduces the volumetric flow rate required as compared to a pure gas system. The reduced flow rate, in turn, leads to a high, equilibrium partial pressure of tritium in the primary coolant. If the tritium recovery system efficiency is 100% and all of tritium remains in the gas phase, then the equilibrium partial pressure of tritium is 100 *Pa*. A tritium partial pressure of 100 *Pa* requires careful consideration of containment problems and blanket tritium inventory. The particulates in the coolant may help to reduce the partial pressure by surface adsorption of the tritium.

SUMMARY

During the ARIES-I scoping phase of the ARIES project, many blanket concepts were considered, a ceramic-composite, gas-cooled design was chosen for a detailed design study. The coolant is low pressure CO_2 with a dilute suspension of particulates (either SiC or SiO_2). The particulates greatly enhance the effective heat capacity of the coolant, thus allowing low pressure operation. The structure is a high-strength, silicon carbide composite. High strength and low coolant pressure enable the blanket to be made of few, reasonably large modules. The breeder is Li_2O in the form of plates. No neutron multiplier is required to achieve adequate tritium breeding. The low-activation and low-afterheat properties of the SiC structure provide for the potential of a passively safe design. Further analysis of accident scenarios and tritium systems is on-going.

REFERENCES

Conn, R. W. and Najmabadi, F. (1987). Visions of the Future, A Program in Tokamak Reactor Study. UCLA report number UCLA-PPG-1201.

Fitzer, E. and Gadow, R. (1986). Fiber-Reinforced Silicon Carbide. Am. Ceram. Bull., 65 [2] 326-335.

Hopkins, G. R. and Cheng, E. (1983). Low Activation Fusion Rationale. Nucl. Technol./Fusion, Vol. 4, pp 528-544.

Najmabadi, F. (1988). The ARIES Tokamak Fusion Reactor Study — The Scoping Phase. Proceedings of the 8th Topical Meeting on the Technology of Fusion Energy, Salt Lake City.

Sharafat, S., et al. (1989). Silicon Carbide Composites as Structural Material for the ARIES Reactor Study. These proceedings.

Wong, C. P. C. and Hasan, M. (1989). Thermal-Hydraulic Design of a Solid Particulate Fusion Reactor Blanket. These proceedings.

Hasan, M. Z. and Martin, R. C. (1989). Use of Dissociating Gases as Primary Coolants and Working Fluids in Power Cycles for Fusion Reactors. These proceedings.

Table 1

TABLE-I: Operating Parameters for the ARIES-I Reactor (April 1989).

Fusion power	2350. MW
Total thermal power	2608. MW
First wall area	413. m^2
14 MeV neutron wall load	4.5 MW/m^2
First wall surface heating	0.68 MW/m^2
First wall volumetric heating	30. MW/m^3
Structure material	SiC ceramic-composite
Coolant type	CO_2 w/ particulates
Breeder type	Li_2O w/ SiC cladding
Coolant pressure	0.5 MPa
Inlet temperature	250. °C
Outlet temperature	700. °C
Peak first wall temperature	946. °C
Peak breeder temperature	954. °C
Maximum first wall velocity	43. m/s
Particle size range	5 to 10 μm
Particle volume fraction	1.5 %
Pressure drop	0.017 MPa
Pumping power	1.7 MW
Thermal cycle efficiency, gross	46 %

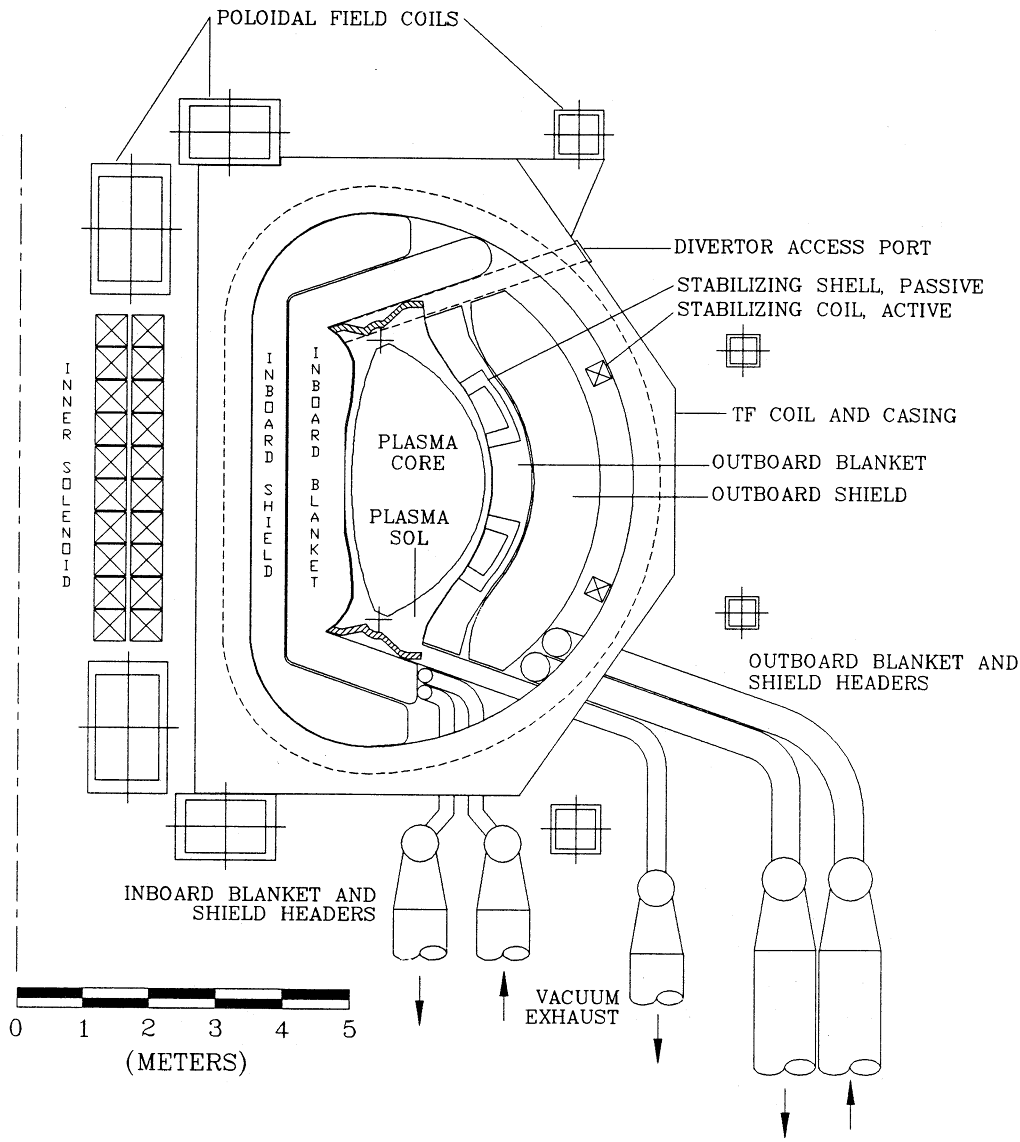

Figure 1. Poloidal cross section of the ARIES-I fusion reactor.