

Reprint from

PLASMA PHYSICS
AND CONTROLLED
NUCLEAR FUSION RESEARCH
1992

PROCEEDINGS OF THE
FOURTEENTH INTERNATIONAL CONFERENCE ON PLASMA PHYSICS
AND CONTROLLED NUCLEAR FUSION RESEARCH
HELD BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
IN WÜRZBURG, 30 SEPTEMBER-7 OCTOBER 1992

In four volumes

VOLUME 3

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 1993

DIRECTIONS FOR ATTRACTIVE TOKAMAK REACTORS: THE ARIES-II AND ARIES-IV SECOND-STABILITY DESIGNS

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Abstract

DIRECTIONS FOR ATTRACTIVE TOKAMAK REACTORS: THE ARIES-II AND ARIES-IV SECOND-STABILITY DESIGNS.

ARIES is a research program to develop several visions of tokamak reactors with enhanced economic, safety, and environmental features. The ARIES study has developed four visions for tokamaks, each having a different degree of extrapolation in physics and technology. All four designs are steady-state, 1000-MWe (net) power reactors. The ARIES-II and ARIES-IV designs assume potential advances in plasma physics, such as second-stability operation, that are predicted by theory but are not yet well established experimentally. The two designs have the same fusion plasma but different fusion-power-core designs. There are only minor differences between the ARIES-II and ARIES-IV plasma parameters. ARIES-IV is a 1000-MWe reactor with an average neutron wall loading of 3 MW/m^2 , and a mass power density of about 120 kWe/tonne of fusion power core. The reactor major radius is 6.1 m, the plasma minor radius is 1.5 m and the plasma elongation is 2, and the plasma triangularity is 0.67. The plasma current is low (6.8 MA), the magnetic field strength on-axis is 7.7 T (corresponding to a maximum field at the coil of 16 T), and the toroidal beta is 3.4% (corresponding to a Troyon coefficient of 6). The plasma operating regime is optimized such that most of the plasma current ($\sim 90\%$) is provided by the self-sustained bootstrap current. The ARIES-II reactor uses liquid lithium as the coolant and tritium breeder. The vanadium alloy, V-5Cr-5Ti, is used as the structural material so that the potential of low-activation metallic blankets can be studied. The ARIES-IV reactor uses helium as the coolant, a solid tritium-breeding material (Li_2O), and silicon carbide (SiC) composite as the structural material. The waste produced by neutron activation in both designs is found to meet the criteria allowing shallow-land burial under U.S. regulations. The cost of electricity for the ARIES-II-IV class of reactors is estimated to be about 20% lower than comparable, steady-state first-stability reactors (e.g. ARIES-I).

1. INTRODUCTION

The ARIES study was undertaken to determine the economic, safety, and environmental potential of steady-state tokamak fusion reactors and to identify physics and technology areas with the highest leverage for achieving attractive fusion power plants [1]. To this end, the ARIES team

Table I
Major Characteristics and Parameters of
ARIES-II and ARIES-IV Second-Stability Tokamak Reactors

A. <u>PLASMA PARAMETERS*</u>		
Plasma major radius (m)		6.1 (5.6)
Plasma minor radius (m)		1.5 (1.4)
Plasma vertical elongation, κ_x		2
Plasma triangularity, δ_x		0.67
Plasma current (MA)		6.8 (6.5)
Toroidal field on axis (T)		7.7 (8.1)
Toroidal field on the coil (T)		16
Toroidal beta		0.034
Poloidal beta		5.4
Electron density ($\times 10^{20} \text{ m}^{-3}$)		2.0 (2.2)
Plasma temperature (keV)		10.
Current-drive method	ICRF fast wave	
	Lower hybrid	
Bootstrap-current fraction		0.9
Current-drive power (MW)		70
Impurity control system	Double-null	
	Gas-target divertors	
Fueling method	Pellet injection	
Net electric output (MW)		1,000

has performed detailed reactor-design work and has explored reactor optimization and trade-off sensitivities using a cost-based systems code. The design effort has been directed to maximize the environmental and safety attributes of fusion through careful design and selection of low-activation material.

Four ARIES designs have been produced, each with a different degree of extrapolation in physics and technology, to yield a spectrum of tokamak reactor concepts based on various potential advances in physics and technology. In turn, this should provide a sensible basis to evaluate the potential of the tokamak as a commercial reactor.

All four ARIES designs are steady-state, 1000-MWe (net) power reactors. The ARIES-I design [2] assumes a minimum extrapolation in physics and, hence, is close to the present-day tokamak data base (*e.g.*, ARIES-I operates in the first MHD stability regime [3]). It incorporates

Table I (Continued)

B. ARIES-II FUSION POWER CORE PARAMETERS

Coolant	Liquid lithium
Structural material	V-5Cr-5Ti
Tritium breeder	Liquid lithium
Neutron multiplier	None
Blanket energy-multiplication ratio	1.4
Tritium breeding ratio	1.1
Coolant inlet temperature (°C)	330
Coolant outlet temperature (°C)	610
Coolant pressure (MPa)	0.5
Average neutron wall loading (MW/m ²)	3.1
Fusion power (MW)	1,950
Thermal power (MW)	2,630
Gross thermal-cycle efficiency	46%
Recirculating power fraction [†]	17%
Net plant efficiency	38%
Mass power density (kWe/tonne of FPC)	100

C. ARIES-IV FUSION POWER CORE PARAMETERS

Coolant	Helium
Structural material	SiC composite
Tritium breeder	Li ₂ O
Neutron multiplier	Be
Blanket energy-multiplication ratio	1.2
Tritium breeding ratio	1.1
Coolant inlet temperature (°C)	350
Coolant outlet temperature (°C)	750
Coolant pressure (MPa)	10
Average neutron wall loading (MW/m ²)	2.9
Fusion power (MW)	2,140
Thermal power (MW)	2,590
Gross thermal-cycle efficiency	49%
Recirculating power fraction [†]	21%
Net plant efficiency	39%
Mass power density (kWe/tonne of FPC)	120

* Most of plasma parameters are similar for the two designs.

If different, the ARIES-II parameters are given in parenthesis.

† Includes 4% designated for auxiliary site power.

technologies that will be available over the next 5 to 20 years, given the existing trends and gradients for these technologies inside and outside of the fusion program. The ARIES-III study focuses on the potential of tokamaks to operate with a D-³He fuel system as an alternative to DT [4].

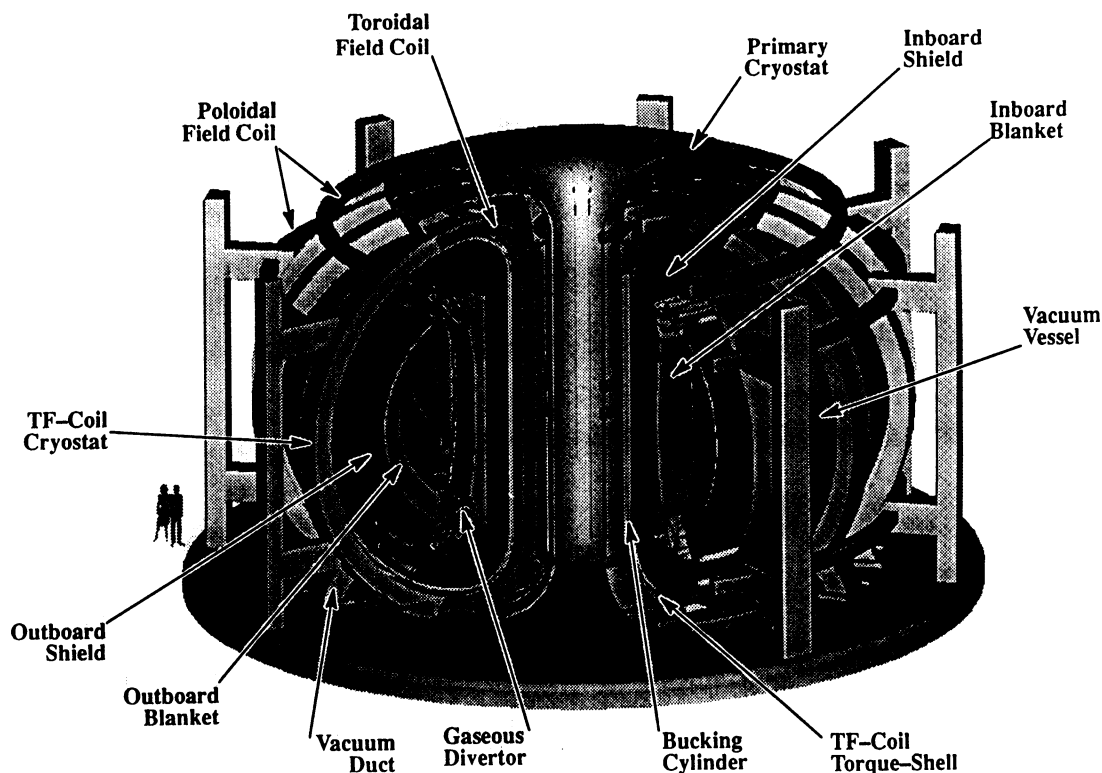


FIG. 1. Elevation view of the ARIES-IV fusion power core.

The ARIES-II and ARIES-IV designs assume potential advances in plasma physics (*e.g.*, second stability operation [5]) that are predicted by theory but are not yet well established experimentally. These two designs have the same fusion plasma but different combinations of structural material and coolants.

The major characteristics and parameters of the ARIES-II and ARIES-IV designs are given in Table I. A cross section of the ARIES-IV fusion power core is shown in Figure 1. There are only minor differences between the ARIES-II and ARIES-IV plasma parameters, due to differences in the engineering design of the fusion power core (blanket energy multiplication, pumping power, thermal cycle efficiency, blanket and shield thickness). ARIES-IV is a 1000-MWe reactor with an average neutron wall loading of 3 MW/m^2 , and a mass power density of about 120 kWe/tonne of fusion power core. The reactor major radius is 6.1 m, the plasma minor radius is 1.5 m, and the plasma elongation is 2, and the plasma triangularity is 0.67. The plasma current is low (6.8 MA), the magnetic field strength on-axis is 7.7 T (corresponding to a maximum field at the coil of 16 T), and the toroidal beta is 3.4% (corresponding

to a Troyon coefficient of 6 [3]). The plasma operating regime is optimized such that most of the plasma current ($\sim 90\%$) is provided by the self-sustained bootstrap current [6].

The ARIES-II reactor uses liquid lithium as the coolant and tritium breeder. The vanadium alloy, V-5Cr-5Ti, is used as the structural material so that the potential of low-activation metallic blankets can be studied. The ARIES-IV reactor uses helium as the coolant, a solid tritium-breeding material (Li_2O), and silicon carbide (SiC) composite as the structural material. The blanket design is similar to that of ARIES-I [2], but several refinements have been incorporated in order to further improve the safety and environmental characteristics. The waste produced by neutron activation in both designs is found to meet the criteria allowing shallow-land burial under U.S. regulations [7]. The cost of electricity for the ARIES-II-IV class of reactors is estimated to be about 20% lower than comparable, steady-state first-stability reactors (*e.g.*, ARIES-I).

In the following sections, we describe the physics and fusion power core engineering of the ARIES-II and ARIES-IV designs. Greater detail can be found in an extensive report of this research [2].

2. PLASMA ENGINEERING

Steady-state operation of tokamaks requires that plasma current be maintained by non-inductive means. The most likely candidates for external current-drive methods (*i.e.*, neutral beam, ICRF fast wave, and lower hybrid) have projected low efficiencies at reactor-relevant plasma conditions [8]. Therefore, the optimum steady-state tokamak reactors operate at plasma regimes in which the total recirculating power is minimized by lowering the plasma current and maximizing the self-induced bootstrap-current fraction [6]. For tokamak reactors operating in the first MHD stability regime [3], such as ARIES-I and SSSTR [9], this optimum regime of operation is at relatively high plasma aspect ratios ($A \sim 4$ to 5), low plasma currents ($I \sim 10$ MA), high poloidal betas ($\epsilon\beta_p \sim 0.6$), and relatively high safety factors, leading to a high bootstrap-current fraction (which is usually limited to $\sim 70\%$ based on self-consistent MHD and bootstrap-current calculations). The choice to operate at lower plasma current in the first MHD stability regime then leads to a low toroidal beta ($\sim 2\%$ to 3%) and the need for high magnetic field in order to achieve an economical fusion power density.

In order to improve the reactor economics, the cost of the magnet system should be reduced (while maintaining or increasing the fusion

power density) through an increase in the plasma beta. Theoretical studies of the second MHD stability regime with higher plasma-beta values have been on-going [5]. Recent analyses [10], including those performed for the ARIES-III second-stability D-³He reactor [4], indicate that operation in second-stability requires an on-axis safety factor of ~ 2 with a corresponding increase in the edge safety factor. Furthermore, since second stability plasmas have a high poloidal beta, the bootstrap current fraction is high and can exceed the equilibrium plasma current. In this case, the current-drive system should drive the seed plasma current as well as canceling the bootstrap current-density overdrive in the outer regions of the plasma, which increases the current-drive power and associated costs. Thus, the optimum second-stability plasma equilibria are those in which the self-sustained bootstrap current density matches the required plasma equilibrium current. These equilibria have only moderate toroidal beta ($\sim 4\%$ to 6%) (corresponding to a Troyon coefficient of 6).

Stability analyses of ARIES-II/-IV-type second-stability equilibria indicate that the plasma β decreases as the aspect ratio is increased (for $A \sim 3$ to 5). However, for a fixed maximum-field strength on the superconducting coils, the on-axis toroidal-field strength increases with increasing aspect ratio. As a result, the fusion power density (and the cost) remains insensitive to the value of aspect ratio (for $A \sim 3$ to 5). Given the engineering advantages of operating at higher aspect ratio and because preliminary indications that start-up at higher values of aspect ratio may be less difficult, a value of $A = 4$ has been adopted for the ARIES-II and ARIES-IV design.

Many second-stability equilibria have been examined for ARIES-II-IV reactors. These equilibria all have modest shaping, smooth pressure and surface-averaged current-density profiles, and a monotonic q profile. It was found that stability to $n = \infty$ ballooning modes requires an on-axis safety factor, $q_0 = 2$, a high edge safety factor ($q^*/q_0 \gtrsim 2.4$), and a high plasma triangularity ($\delta_{95} \gtrsim 0.45$). Unfortunately, none of the examined equilibria was found to be stable to $n = 1$ kink modes with no stabilizing walls. The reference equilibrium is, however, stable to $n = 1$ kink modes if a perfectly conducting wall is located at a radial distance of $0.25a$.

The effect of a finite conductivity wall on the kink stability of the plasma is unknown. For the ARIES-II-IV designs, it is assumed that the plasma rotation would compensate for finite wall conductivity. (An analogy can be drawn with the problem of a plasma ring which is translating inside a cylindrical wall with finite conductivity. The ring can

be stable if the plasma transit time is shorter than the wall skin time). Assuming a plasma rotational velocity of 50 km/s, the conducting wall for ARIES-II design (made of vanadium alloy) should be 7.5 cm thick. For the ARIES-IV design, a 2-cm thick zone of beryllium would be sufficient. These materials are present in both fusion power core designs and are located closer than $0.25*a$ to the plasma.

The average plasma ion temperature is 10 keV (central temperature of 26 keV) and the average ion density is $2 \times 10^{20} \text{ m}^{-3}$. About 90% of plasma current (6.8 MA) is provided by the bootstrap current. A central seed current of 0.6 MA is needed and is driven by ICRF fast waves [11] (19 MW at a frequency of 124 MHz with an efficiency of 23 mA/W). The bootstrap current density slightly exceeds the equilibrium current density in the middle of the plasma. The overdrive current in this region is 0.18 MA and lower hybrid waves are utilized for canceling the overdrive current in this region (12 MW at a frequency of 8 GHz with an efficiency of 11 mA/W). Because of collisional effects, the bootstrap current density decreases rapidly in the cold plasma edge. A total of 0.16 MA of current is supplied to this region by lower hybrid waves (39 MW at a frequency of 8 GHz with an efficiency of 3 mA/W).

The required energy confinement time is about 3 times the prediction of the ITER-89P L-mode [12] model. This value is comparable to those of VH-mode discharges in DIII-D and high- ℓ_i discharges both in TFTR and DIII-D. About 20% of plasma energy is radiated mainly as bremsstrahlung. An ash-particle to energy confinement ratio $\tau_p/\tau_E = 9$ is assumed. The alpha particle density is estimated to be $2 \times 10^{19} \text{ m}^{-3}$, corresponding to a burn-up rate of 18%. Loss of fast alpha particles due to stochastic ripple effect or prompt orbit loss has been analyzed and found to be negligible (the toroidal-field ripple is 0.5% on the outboard plasma edge).

Because of the high triangularity needed for second-stability operation, single-null divertors do not appear to be feasible. The ARIES-II-IV designs, therefore, utilize double-null divertor configurations. The reference designs incorporated gas-target divertors in which the plasma is terminated in a high-density gas [13]. Preliminary analyses indicate that the plasma temperature can be reduced to about 2 eV and the plasma energy is deposited more-or-less uniformly on the side wall of the divertor chamber (with a peaking factor of $\lesssim 1.5$). The average heat load on the outboard divertor wall is estimated to be $\sim 2.5 \text{ MW/m}^2$. Neutral-transport studies show that the leakage of divertor gas target into the main plasma is negligible. Because of low core-plasma temperature and

high density, a conventional divertor appears to be feasible using the same structural material and coolant for ARIES-II and ARIES-IV design. The surface of these divertors, however, should be coated with tungsten to limit sputtering erosion.

Plasma start-up is an important issue for second-stability plasma since plasma evolution to its final steady-state burn parameters should follow a stable access path to the second-stability regime. A preliminary start-up scenario has been identified which uses the current drive system both for heating and current drive. The total heating power required for ignition is about 90 MW, and, thus, additional 20 MW of auxiliary heating equipment may be needed for start-up. The current-drive power, however, appears to be adequate for current profile tailoring.

3. FUSION POWER CORE ENGINEERING

The ARIES-II-IV fusion power core is divided into 16 sectors (Figure 2). An entire sector is replaced as a single unit during a maintenance

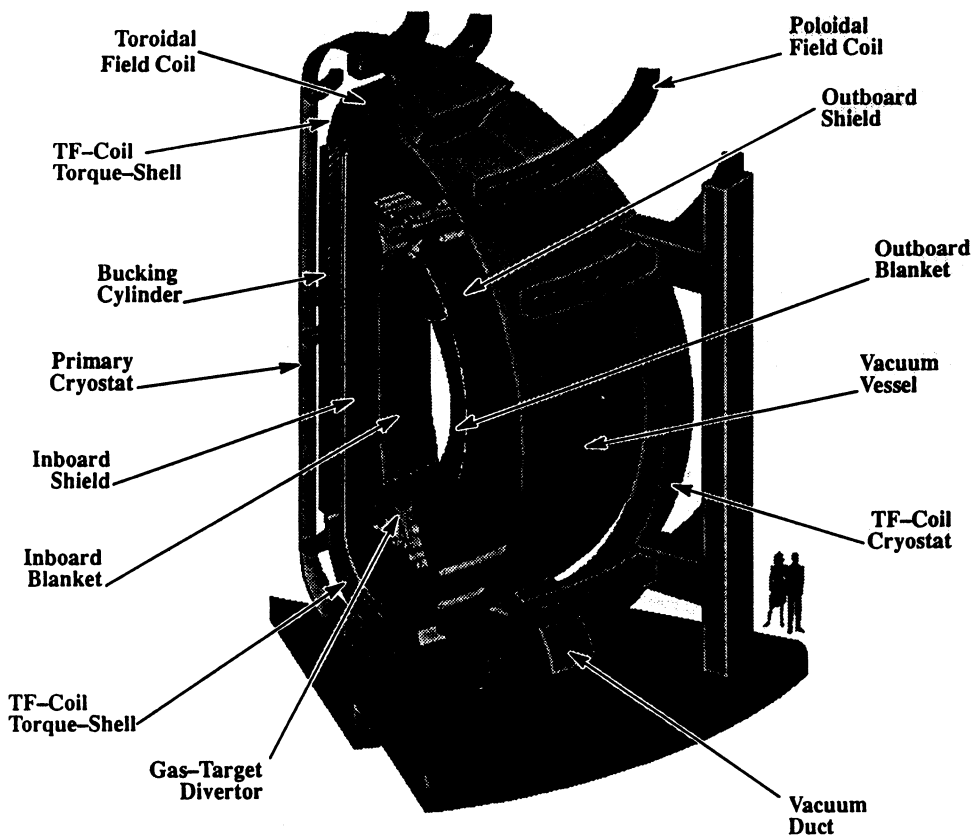


FIG. 2. Elevation view of one of the ARIES-IV fusion power core sectors.

operation. This approach is preferable to a piece-by-piece scheme in which components are removed sequentially through the space in between the TF coils (such as that proposed for ITER [12]) because it requires fewer connects/disconnects and joints. In ARIES-II-IV designs, all of the toroidal-field coils, most of the poloidal-field coils, and the associated support structures are located in a cryostat as shown in Figure 1. These coils remain in place during the maintenance operation. The outer leg of the toroidal-field coils is extended by 1.2 m so that a “pie-shaped” sector can be horizontally removed and replaced through the space between the TF coils.

The ARIES-II-IV toroidal-field (TF) coils represent a modest extension of present technology since the maximum field on the ARIES-II-IV toroidal-field (TF) coils is 16 T. The magnetic stored energy, however, is large because of the extended outer leg of the TF coils. The winding concept is similar to that of ARIES-III [4] and of a new concept proposed for ITER. This design has a very high capability to carry out-of-plane loads and, thus, no structure is needed in the outboard region of the torus, facilitating access for maintenance. Second-stability reactors require a lower toroidal-field ripple in order to maintain fast- α -particle losses to an acceptable level (due to stochastic ripple effect or prompt orbit loss). The enlarged outer leg of the toroidal-field coils in ARIES-II-IV designs reduces the toroidal-field ripple on the outboard plasma surface to $\sim 0.5\%$ —fast α particle losses are negligible for this case.

3.1 ARIES-II Fusion Power Core

The ARIES-II fusion power core uses V-5Cr-5Ti vanadium alloy as the structural material and liquid lithium as the coolant and tritium breeder. The key design issue for the self-cooled liquid-metal blanket is the MHD pressure drop. Because of the relatively high magnetic-field strength at the inboard blanket location of the ARIES-II design, it appears that the MHD pressure drop is large and even exceeds the design-stress limit of the structural material. The ARIES-II fusion-power-core design assumes that a V-Ti-N coating will be formed on the blanket coolant-channel walls in order to reduce the MHD pressure drop. Several liquid-metal loop experiments [14-16] with liquid lithium and vanadium have reported the formation of this V-Ti-N coating on the V-Ti alloy surface. This coating is formed by the interaction of the nitrogen impurity in the liquid lithium with the alloy. The coating appears to be stable and self-healing. Long-term stability and electrical resistivity of this coating in a radiation environment is the critical R&D issue for the ARIES-II fusion power core.

Based on the electrical resistivity of this coating, which was recently measured at Argonne National Laboratory [17] to be about $\sim 5 \times 10^{-5} \Omega\text{m}$, it is estimated that the MHD pressure drop can be reduced by a factor of 15 when compared to a similar design with uninsulated channel walls. Because the MHD pressure drop is reduced substantially by this coating, the ARIES-II blanket design has aimed at simplicity and optimum heat-transfer capability. The first-wall and blanket coolant channels are rectangular with cross sections of 25×50 mm and 55×50 mm, respectively. The channel wall thickness is 3 mm. The coolant inlet pressure is 0.5 MPa and the total MHD pressure drop is estimated at 0.3 MPa. The coolant inlet and outlet temperatures are, respectively, 330 and 610 °C and the maximum temperature of vanadium structure is 660 °C. Tritium recovery is based on a molten salt extraction technique and the tritium inventory in the primary lithium coolant is estimated at 140 g.

The total blanket and shield thicknesses in the inboard and outboard regions are, respectively, 1.1 and 1.6 m. The blanket is 0.2-m thick at the inboard, 0.5-m thick at the outboard, and contains 90% natural lithium and 10% vanadium structural material. The reflector and shield regions use vanadium-alloy structural material with stainless steel and B_4C fillers and are cooled by liquid lithium. The blanket energy-multiplication ratio is estimated at 1.38 and the tritium breeding ratio is 1.1. The end-of-life fluence for the vanadium alloy is estimated to be $16 \text{ MW}\cdot\text{y}/\text{m}^2$ (based on degradation of mechanical properties by He embrittlement process). Thus, the shield lifetime is longer than 30 full power years.

Activation analyses indicate that the first wall, blanket, and shield qualify as Class-C shallow-land burial waste under U.S. regulations [7]. Loss of coolant analysis has been performed and the maximum structural temperature is estimated at 850 °C. The safety consequences of lithium fires are under study. Preliminary indications are that the ARIES-II reactor meets the requirements for passive safety (Level 3) as defined by ESECOM [18,19]

The ARIES-II reactor includes an intermediate coolant loop (with liquid lithium). An advanced steam power cycle, similar to those proposed for coal-burning power plants [20] is used for ARIES-II. The gross thermal efficiency is estimated at 46%.

3.2 ARIES-IV Fusion Power Core

Because of excellent safety and environmental features of the ARIES-I fusion power core [2], a similar design has been adopted for the ARIES-IV reactor (SiC-composite structural material and helium coolant). Several refinements, however, have been incorporated in order to further improve the safety and environmental characteristics. These include replacing the Li_2ZrO_3 tritium breeder of ARIES-IV with low-activation Li_2O and eliminating the tungsten coating of the divertor plates.

The ARIES-IV fusion power core employs a silicon-carbide-composite structural material to be manufactured as a large integrated piece utilizing techniques already in use or under development in the aerospace industry [21-25]. This composite retains many of the desirable features of bulk SiC ceramic, but the addition of SiC fibers greatly reduces the brittleness of the material and produces a high fracture toughness. The increase in toughness creates more freedom in engineering design and allows both tensile and compressive stress in the composites. Desirable features of SiC composite include high-temperature capability, high strength, extensive resource availability, and potentially good resistance to radiation damage. In addition, because the levels of induced activation and after-heat in SiC are quite low, the safety and environmental features of the reactor are enhanced.

The fusion-power-core design is based on a nested-shell configuration, with the coolant flowing through the coolant channels that are embedded in the shells. Each SiC-composite shell is manufactured as a single unit utilizing techniques already developed or under development by the aerospace industry. A sphere-pac of beryllium is located between the first two shells (total Be thickness is 4 cm). A sphere-pac of lithium-oxide (Li_2O) tritium breeder is located between the rest of the blanket shells. Tritium recovery is by a slow-flowing, low-pressure purge stream of helium between the shells. The reflector and shield components are also in the form of nested shells with the appropriate filler materials located between the shells.

The fusion power core is cooled by helium at 10 MPa. The coolant-flow configuration is in the poloidal direction, the coolant enters at the bottom of each shell and flows upwards. The coolant inlet and outlet temperatures are, respectively, 350 and 750 °C. The maximum temperature of the first-wall structural material is 950 °C. The blanket pumping power is 20 MW. An advanced Rankine power cycle (similar to that of ARIES-I) is adopted because the coolant outlet temperature is

sufficiently high, and because this cycle is planned for near-term, coal-fired power plants [20].

Neutronics analyses indicate that the total blanket and shield thicknesses in the inboard and outboard regions are, respectively, 125 and 160 cm. The blanket is 35-cm thick at the inboard and 60-cm thick at the outboard. Because of beryllium safety concerns (tritium production, reaction with air and water, and chemical toxicity), beryllium resource limitations, and high cost of beryllium, tritium breeding capability of the ARIES-IV blanket without any beryllium was considered. The “simulated” three-dimensional tritium breeding ratio was found to be ~ 1.05 . It was estimated that a tritium breeding ratio of 1.1 is required to ensure adequate tritium breeding due to uncertainties in the nuclear data base and the differences between the neutronics model and actual blanket configuration. Therefore, a 4 cm beryllium zone is added to the front of the blanket (the total beryllium inventory is ~ 20 tonnes). The estimated tritium breeding ratio for the ARIES-IV reference design is 1.1 and the blanket energy-multiplication ratio is estimated at 1.18.

The reflector and shield regions also use SiC-composite structural material with SiC bulk material and B_4C fillers. The end-of-life fluence for the SiC composites is estimated to be $13 \text{ MW}\cdot\text{y}/\text{m}^2$ (based on 3% burn-up limit). Thus, the shield lifetime is longer than 30 full power years. Activation analyses indicate that all fusion-power-core materials qualify as Class-C shallow-land burial waste under U.S. regulations [8]. Detailed analyses indicated that the ARIES-I design is passively safe (Level 2) as defined by ESECOM [18,19]. Because the major safety concern of the ARIES-I fusion power core (Li_2ZrO_3 breeder, tungsten coating of the divertor plate) is eliminated in the ARIES-IV design, ARIES-IV is expected to have better safety characteristics. Analyses are on-going to determine if ARIES-IV is inherently safe. This improvement in safety characteristics is expected to reduce the cost of electricity by 15%.

4. SUMMARY

The ARIES-II and ARIES-IV designs assume potential advances in plasma physics (*e.g.*, second-stability operation) that are predicted by theory but are not yet well established experimentally. These two designs have the same fusion plasma but different combinations of structural material and coolants. Analyses indicate that the optimum second-stability plasma equilibria are those in which the self-sustained bootstrap current density matches the required plasma equilibrium current. These

equilibria have only moderate toroidal beta ($\sim 4\%$ to 6%) (corresponding to a Troyon coefficient of 6). The improved physics performance of the ARIES-II-IV fusion power core are utilized both to lower the cost of electricity (by $\sim 20\%$) and to reduce the technological requirements (*e.g.*, lower peak field on the coil, extended outer leg of TF coils to ease maintenance).

The physics of second-stability plasmas is yet to be explored even though experimental indications are encouraging. Although the ARIES-II-IV equilibria are robust for minor changes in the pressure and current density profile, the detail matching of bootstrap current density to the equilibrium current density requires careful control of plasma pressure and driven current profiles. Particle and energy transport in these plasmas are also unknown. Lastly, kink stability of second-stable plasma should be examined. Steady-state or long pulse (much longer than current skin time) experiments are required to thoroughly investigate the trade-offs of second-stability plasma and determine if their potential advantages, as highlighted by ARIES-II-IV research, can be realized.

As with other ARIES designs, the ARIES-II-IV design effort has been directed to maximize the environmental and safety attributes of fusion through careful design and selection of low-activation material. However, extensive fusion nuclear technology and material R&D are required so that the ARIES-II and ARIES-IV fusion power cores are realized. Because of the long lead time, this R&D (including the operation of a 14 MeV neutron source) should start as soon as possible so that these technologies would be available for the fusion demonstration power plant.

ACKNOWLEDGMENT

This research has been supported by the United States Department of Energy, Office of Fusion Energy. Institutions participating in the ARIES study, in addition to UCLA, are Argonne National Laboratory, General Atomics, Idaho National Engineering Laboratory, Los Alamos National Laboratory, Massachusetts Institute of Technology, Oak Ridge National Laboratory, Princeton Plasma Physics Laboratory, Rensselaer Polytechnic Institute, University of Wisconsin, Georgia Institute of Technology, Culham Laboratory, and Japan Atomic Energy Research Institute.

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DISCUSSION

F. ENGELMANN: Is there any analysis on how to get round the problem of instabilities caused by the presence of fast (fusion generated) ions in a tokamak reactor working with D-³He?

F. NAJMABADI: Not to my knowledge.