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Recent Designs for Advanced Fusion Reactor Blankets*

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ABSTRACT

A series of reactor design studies based on the Tokamak configuration have been carried out under the direction of Professor Robert Conn of UCLA. They are called ARIES-I through IV and PULSAR I and II. The key mission of these studies is to evaluate the attractiveness of fusion assuming different degrees of advancement in either physics or engineering development. Also, the requirements of engineering and physics systems for a pulsed reactor were evaluated by the PULSAR design studies. This paper discusses the directions and conclusions of the blanket and related engineering systems for those design studies.

ARIES-I investigated the use of SiC composite as the structural material to increase the blanket temperature and reduce the blanket activation. Li_2ZrO_3 was used as the breeding material due to its high temperature stability and good tritium recovery characteristics. To reduce the activation caused by the using of Zr, isotopic tailoring is required. Also, W was selected as the divertor target. The activation caused by the Zr and W, even with isotopic tailoring, reduced the safety advantage for the SiC blanket.

The ARIES-IV is a modification of ARIES-I. The plasma was in the second stability regime. Li_2O was used as the breeding material to remove Zr. A gaseous divertor was used to replace the conventional divertor so that high Z divertor target is not required. We investigated the possibility of breeding without the use of Be. However, tritium self sufficiency could not be assured with the uncertainties in the neutronic data. The safety advantage of ARIES-IV was enhanced by the removal of the high activation materials.

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The physics of ARIES-II was the same as ARIES-IV. The engineering design of the ARIES-II was based on a self-cooled lithium blanket with a V-alloy as the structural material. Even though it was assumed that the plasma was in the second stability regime, the plasma beta was still rather low (3.4%). To achieve an acceptable neutron wall loading, the magnetic field is rather high. This put extra burden on a self-cooled liquid metal blanket. It was determined that a self-cooled lithium blanket with bare walls was not acceptable for a reactor with ARIES-II type parameters. Therefore, an insulating coating is required to assure an acceptable design window to reduce the MHD pressure drop.

The ARIES-III is an advanced fuel (D-^3He) tokamak reactor. The reactor design assumed major advancement on the physics, with a plasma beta of 23.9%. A conventional structural material is acceptable due to the low neutron wall loading. From the radiation damage point of view, the first wall can last the life of the reactor, which is expected to be a major advantage from the engineering design and waste disposal point of view. Organic coolant was selected as the reactor coolant to reduce the operating temperature compared to He, and to reduce the coolant pressure and thermal efficiency compared to water. However, the use of organic coolant raised safety and decomposition concerns.

The PULSAR I and II designs are based on the ARIES-IV and II designs to evaluate the issues associated with a pulsed reactor. The key goal is to make a comparison between SS and pulsed reactors. The key areas of investigation are:

1. Fatigue issues associated with the first wall/blanket/shield and divertor.

2. Evaluation of the operating of the power conversion equipment for a pulsed reactor.

3. The selection and design of a thermal energy storage system for the dwell period.

4. Assessment of disruption effect to the blanket and divertor.

This work is still in progress.

I. INTRODUCTION

The goal of the ARIES project is to investigate the attractiveness of a tokamak as a commercial fusion power plant. Also, the ARIES project will assess the R & D requirements for high payoff areas of fusion research that could lead to significant improvements in the overall performance of tokamak power plants. To achieve this goal, a range of tokamak reactor concepts was considered by the ARIES team in a series study identified as ARIES-I through ARIES-IV.[1-4] Each study assumes different extrapolation on the development of physics and/or engineering from today's data basis. Figure 1 shows the regime of the extrapolation considered, together with the location of each design study. This engineering-physics matrix depicted on Figure 1 was generated at the beginning of the ARIES project to establish the scope and goals of each of the ARIES designs.

The parameters of the four ARIES studies are summarized on Table 1. The plasma of ARIES-I is in the first stable regime. Thus the plasma beta is rather low. This requires high magnetic field. Even with this high field, the neutron wall loading is still rather low, which results in high COE. The physics extrapolations of ARIES-II and IV are much more aggressive. The beta is higher. However, due to the requirements of plasma stability and current drive requirements, the wall loading is still modest. The ARIES-III investigated the attractiveness of D-^{3}He reactor. The requirements from physics is even more aggressive than the ARIES-II and IV.

This paper discussed the blanket designs of the ARIES study. There are three categories of the blanket which are most attractive from safety and environmental point of view.[5] They are:

1. He-cooled SiC blanket.

2. Li-cooled V-alloy blanket.

3. Blanket and shield system for a D-^{3}He reactor.

To push the safety and environmental attractiveness of fusion, it is not surprise that the selection of the ARIES blanket systems end with the same three design which are outstanding from safety and environment considerations.

Due to the limited space of this paper, I will only emphasize the design goal of the blanket, with limited discussion on the configuration. The change of the design goals as the design process progressing will be discussed. The critical issues associated with each design will be summarized.

II. ARIES-I DESIGN

A. Design Goal

The design goal of the ARIES-I was to push the limit of safety and environmental advantage of fusion. Thus, SiC composite was selected as the structural material. For its high temperature stability and low activation, Li_4SiO_4 was selected as the breeding material. To assure tritium breeding requirement, Be was added for neutron multiplier. To increase volumetric heat capacity of the gas coolant without excess pressure, solid particulates suspended in He gas was selected as the coolant.

It is anticipated that this blanket will have high thermal converting efficiency, low activation, and sufficient breeding.

B. Finding of the Design

There are indications that SiC can be an attractive structural material for a fusion reactor. Due to the lack of material data, it is difficult to make final assessment. However, we did not find any critical flaw of the material. Assuming the data base can be developed favorably, SiC composite can push the safety advantage of fusion to a high level, however, the environmental advantage of SiC is not clearly defined. The formation of ^{26}Al may cause considerable problems with waste disposal and/or material recycling.

The solid particulates coolant had some problems. At the velocity we assumed, the estimated erosion rate was very high. There were some indications that the erosion of the structural material by particulates is a threshold reaction, i.e., the erosion rate drops off rapidly as the particle size is reduced below a certain level. However, existing data can not quantify the reduction of the erosion rate for small size particles. Thus, high pressure helium (10 MPa) was selected as the coolant. The pumping power required and the heat transfer both improve with the increasing of the pressure of the helium. The helium pressure was limited to 10 MPa due to the mechanical considerations of the system.

The issue associated with Li_4SiO_4 is the effect of lithium burn up. As the concentration of lithium is reduced, due to the tritium breeding, the chemistry of the material changes. There is a eutectic composition with a melting temperature of about 1000C. Thus, the breeding material temperature is very close to the melting temperature of the material at the eutectic composition. The effect of this high temperature to the tritium recovery is uncertain. For this reason, Li_4SiO_4 was replaced by Li_2ZrO_3 as the breeding material.

To achieve sufficient breeding ratio with Li_2ZrO_3, Be had to be added for neutron multiplication. For the divertor erosion concerns, W coating was used. The use of W, Zr and Be added safety concern to the reactor. To reduce activation, both Zr and W were isotopically tailored.

A supercritical steam cycle was selected as the power conversion system, with a converting efficiency of 49%. The configuration of ARIES-I blanket is shown on Fig. 2.

C. Key Issues of the ARIES-I Design

The material data basis of SiC for fusion applications is very scarce. Even the life time of the material under irradiation can not be estimated at this time. Therefore, the material data basis have to be developed to assess the performance of SiC composite for fusion applications.

The dominate activation of the blanket came from W and Zr, even with isotope tailoring. Thus, it is important to reduce or eliminate the usage of those material to realize the safety advantage of SiC. Be is also a safety concern due to its reactivity with the water. Also, the use of the Be causes peak nuclear heating in the breeding material, which causes engineering problems in the blanket design.

III. ARIES-IV DESIGN

A. Design Goal

The ARIES-IV design is a fellow-up to the ARIES-I design. Thus, it is more logical to discuss the ARIES-IV design follow the ARIES-I discussion.

At the end of the ARIES-I design, a discuss was held within the ARIES team regrind how to improve the SiC design for fusion. The logic of the ARIES-IV was to remove some of the problems identified from the conclusion of the ARIES-I.

- Replace of Li_2ZrO_3 by Li_2O.
- Remove Be.
- Remove W from the divertor regime.

Also, the plasma was assumed to be in the second stability regime. Thus, the neutron wall loading was higher than that of ARIES-I.

The configuration of the ARIES-IV blanket is shown on Figure 3.

B. Finding of the design:

Neutronics calculation has shown that, even with Li_2O as the breeding material, tritium self-sufficiency can not be assured without the use of Be. However, the amount of Be required was reduced comparing to ARIES-I. The safety concerns, and the heating in the breeding material, were all reduced. The calculated breeding ratio without Be was sufficient for tritium self-sufficiency, but can not cover the uncertainties on the data basis and model errors.

For a plasma in the second stability regime, a highly conducting shell is required for controlling of kink stability. This conducting shell was to be within 20 cm of the first wall. This was a difficult requirement for the SiC design. Any material added would have severe impact on the activation, after heat, and cause temperature limitations. For the ARIES-IV design, the Be layer was assumed to be able to maintain as a conducting layer. However, whether this can be done in the radiation environment and thermal cycling conditions is very questionable.

With no Zr or W in the blanket and divertor regime, the safety characteristics of the reactor improved. The activation and afterheat level were all very low. Thus the highest safety rating can be reached. However, the waste disposal rating was still handicapped by the production of ^{26}Al, as was the case of ARIES-I.

C. Key Issues of the ARIES-IV Design

The requirement of the data base of SiC composite is same as that discussed for the ARIES-I.

The requirement of a conducting shell within 20 cm of the first wall is a critical issue for the ARIES-IV design. The ability of the Be layer to provide the conducting requirement over long time period can not be assured. This problems can only be resolved by the combined research effort from physics, material scientists and reactor engineers.

IV. ARIES-II DESIGN

A. Design Goal

The design goal of the ARIES-II is to evaluate a self-cooled liquid metal blanket for a high beta tokamak (second stability). If the plasma beta is high, the required magnetic field would be low or modest. The lower magnetic field would make the engineering problems associated with a self-cooled liquid metal blanket more manageable.

A self-cooled liquid metal blanket has the capability of handling higher heat flux than a He-cooled system. A plasma in the second stability regime has the capacity to provide higher neutron wall loading. Thus, we intend to design a reactor system with a high power density to take full advantages of lithium-cooled first wall and divertor.

To design a reactor with high power density, high temperature, and good safety characteristics, V-alloy was selected as the structural material.

The configuration of the ARIES-II design is shown on Figure 4.

B. Finding of the Design

A very important finding of the ARIES-II physics was that the second stable physics can only provide a moderate plasma beta, as shown on Table 1. Thus, the neutron wall loading was modest, and the required magnetic field was high. Under those condition, the MHD pressure drop was very high. In fact, for the IB of the ARIES-II blanket, there was no design window space, unless an insulating coating was used to reduce the liquid metal MHD pressure drop.[6]

A TiN coating was observed on corrosion experiments between Li and V/Ti alloys. Thus, this coating was used to provide the required insulating functions to reduce the MHD pressure drop. Some very preliminary measurement of the resistance of this coating was carried out, and the effect of this resistance to the MHD pressure drop was estimated. Based on this work, it was concluded that the MHD pressure drop can be reduced by about a factor of 10. Thus, the MHD pressure drop became modest.

Both the V-alloy and lithium coolant can go to very high temperature. Thus, the coolant exit temperature reaches 600C. A high temperature steam cycle was used for power conversion. An conversion efficiency of 46% was achieved.

C. Critical Issues of the ARIES-II

More material data for the V-alloy is required. This include the radiation effect, the interstitial impurity effect, and others. Although there are much more material data for V-alloy than SiC composite, but that is still not sufficient.

The development of the insulating coating to reduce the MHD pressure drop is a key R/D issue for a self cooled liquid metal blanket. This coating has to be reliable under fusion environment. Thus, it will subject to neutron irradiation, high heat flux, and thermal cycling.

Tritium recovery from lithium remains to be a key technical issue. To recovery tritium from lithium to a concentration of ~ 1 appm is a very difficult task, due to the high solubility of tritium in the lithium.

V. ARIES-III DESIGN

A. Design Goal

The ARIES-III was to evaluate the engineering issues and the safety characteristics of a fusion power plant with D-^{3}He plasma. Since neutron yield is very low, it is expected that the engineering design will be simple, and the safety characteristics of the plant can be much more attractive than that of a D-T plant.

To achieve the potential of a D-^{3}He power plant, direct convert power conversion systems should be evaluated.

B. Findings of the Design

From physics considerations, the plasma was selected in the second stability regime. After extensive evaluation, it was concluded that no direct conversion system would be consistent with the restrictions from the physics.[7] Thus, a thermal conversion system was adopted for the ARIES-III.

From power balance point of view, a highly reflective first wall was required for the ARIES-III. The reference material selected was Be. However, the selection of Be limited the first wall temperature to about 650C. This requirement put severe limitation on the power conversion efficiency.

Also, for a D-^{3}He reactor, the first wall heat load was very high. Therefore, a coolant with high heat transfer capability is required. Also, the recirculation power of the plasma is very high (23%). Thus, a coolant with high thermal converting efficiency was required. Thus, the coolant selected had to fulfill the following conditions:

a. Operating under a limited Tmax,

b. Have a good heat transfer capability, and

c. Have a good thermal converting potential.

To satisfy conflicting requirements, an organic coolant was selected for the ARIES-III reactor.

Due to the low neutron yield, a low activation steel was used as the structural material. The radiation damage to the structure was low so that the first wall was expected to be a lifetime component.

C. Critical Issues of ARIES-III

The first wall reflectivity requirement of a D-^{3}He is very stringent. It is important to calculate the globe reflectivity based on the local reflectivity and the geometry of the reactor. This reflectivity has to be very high from the plasma power balance point of view. Also, the effect of irradiation to the reflectivity has to be assessed.

The D-^{3}He had a large stored energy in the plasma. Preliminary calculation indicated that the first wall and the divertor can service one disruption. More detailed calculation to assess the disruption effects have to be carried out.

The organic coolant has some characteristics that are very suitable for a D-^{3}He reactor, but there are some safety concerns due to its reactivity with oxygen. Other alternative coolants have to be considered.

VI. CONCLUSIONS

The economic, safety and environmental attractiveness of nuclear fusion can be realized by different combination of physics, engineers, fuel cycle and material combinations. The ARIES study evaluated the attractiveness of fusion using different combinations of aggressiveness of extrapolation from today's data base. The ARIES designs have been selected as the ones most likely to realize the potential of fusion. A key problem in the evaluation of the blanket system is the lack of material and engineering data on the material selected. Therefore, it is difficult to assess the feasibility of the engineering system described. For fusion development, material research has to be a key element on the R & D program.

SiC is an attractive material for fusion applications, but also has the least data available. Even the life time of SiC can not be documented at this time. However, SiC has low activation, low after heat, high temperature capability, and abundant resource. Material research on SiC has to be carried out to assure that SiC is compatible to fusion environment.

For a self-cooled liquid metal blanket, the development of insulating coating to reduce the MHD pressure drop is a key issue. This insulating coating has to be reliable over long period of operation. Thus, the ability of self-healing will be very desirable.

The engineering for the D-^{3}He design that is associated with heat transfer, power conversion, and the first wall requirement on the reflectivity present a difficult problem. Direct conversion is very desirable to reach the potential of a D-^{3}He fusion. Unfortunately, tokamak is not compatible with direct conversion.

The requirement of the conducting shell to maintain plasma stability is a difficult design problem for a power reactor. This problem may not be severe for a metallic blanket, if the conductance provided by the blanket is sufficient for the plasma requirement. This is a particular difficult problem for the ceramic design. The addition of the conducting shell is not consistent with the low activation and high temperature characteristics of a ceramic blanket.

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Table 1. ARIES Parameters

Parameter	ARIES-I	ARIES-II	ARIES-III	ARIES-IV
Fuel	D-T	D-T	D-^{3}He	D-T
Aspect ratio, A	4.5	4.0	3.0	4.0
Major radius, R_T (m)	6.75	5.16	7.50	5.80
Minor radius, a_p (m)	1.50	1.29	2.50	1.45
Stability parameter, $\varepsilon\beta_p$	0.6	1.6	1.8	1.6
Toroidal beta, β(%)	1.9	3.4	23.9	3.4
On-axis magnetic field, β_T (T)	11.3	7.7	7.6	7.3
Coil magnetic field, B_C (T)	21.3	16.0	14.0	16.0
Poloidal beta, β_p	2.8	4.9	5.4	4.9
Plasma current, I_ϕ (MA)	10.3	5.3	29.9	5.6
Bootstrap fraction f_{BC}	0.68	1.1	1.25	1.1
Current-drive power, P_{CD} (MW)	98	27	172	30
Thermal efficiency, η_{TH}	0.49	0.45	0.44	0.49
Engineering gain Q_E	5.0	11.8	4.1	7.9
Mass power density, (kWe/tonne)	85	144	90	164
Neutron wall loading, I_w (MW/m^2)	2.5	3.5	0.08	3.2
Level of safety assurance, LSA	2	2	2	1

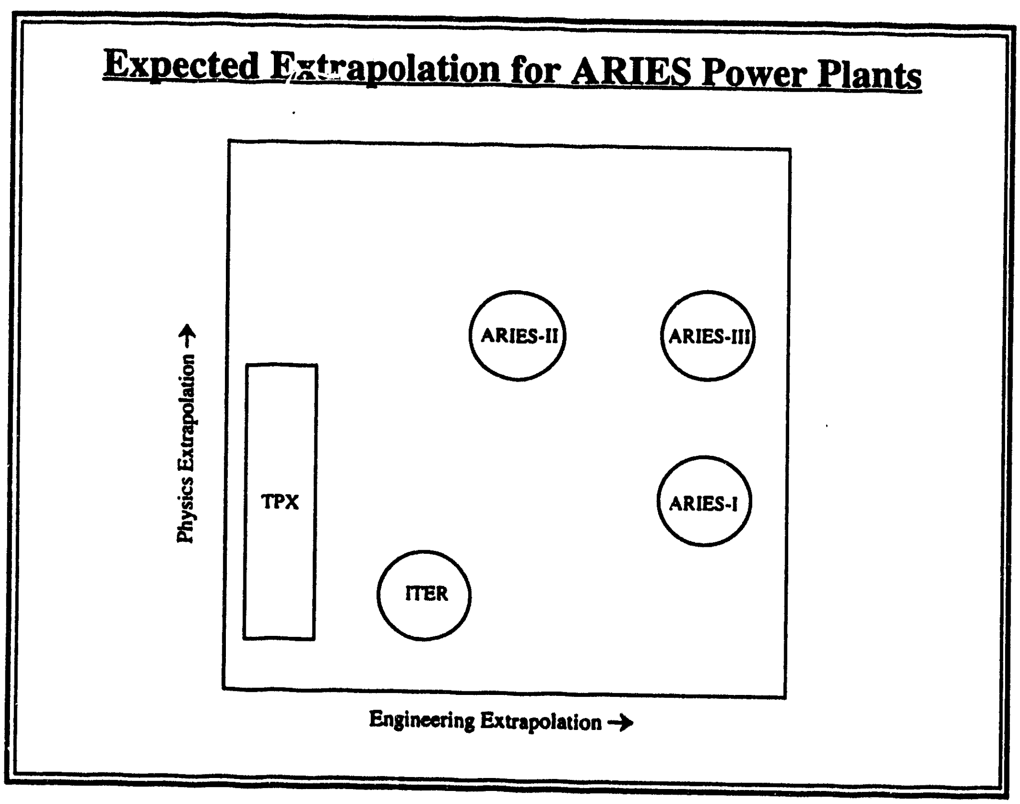

Figure 1

ARIES-I REFERENCE NESTED SHELL BLANKET POLOIDAL

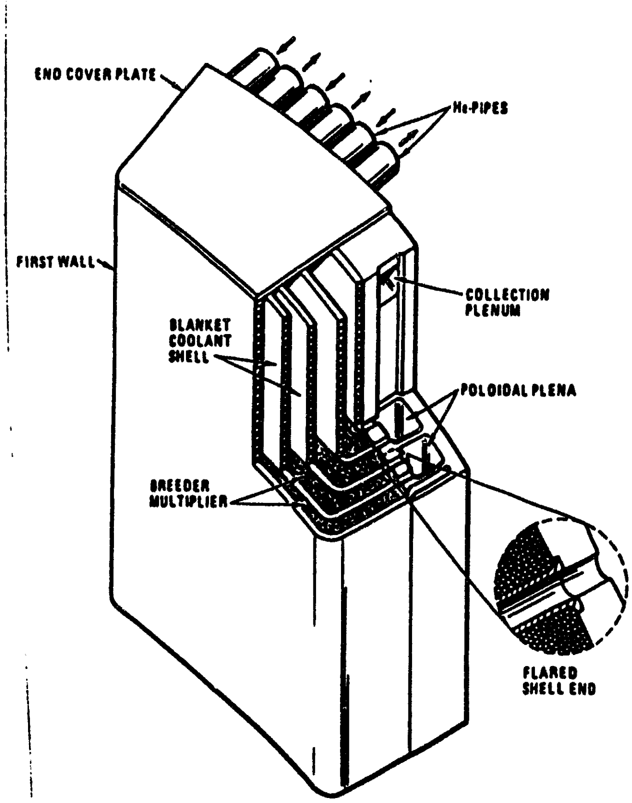

Figure 2

ARIES-IV ALTERNATE BLANKET DESIGN MODULE ASSEMBLY SCHEMATIC

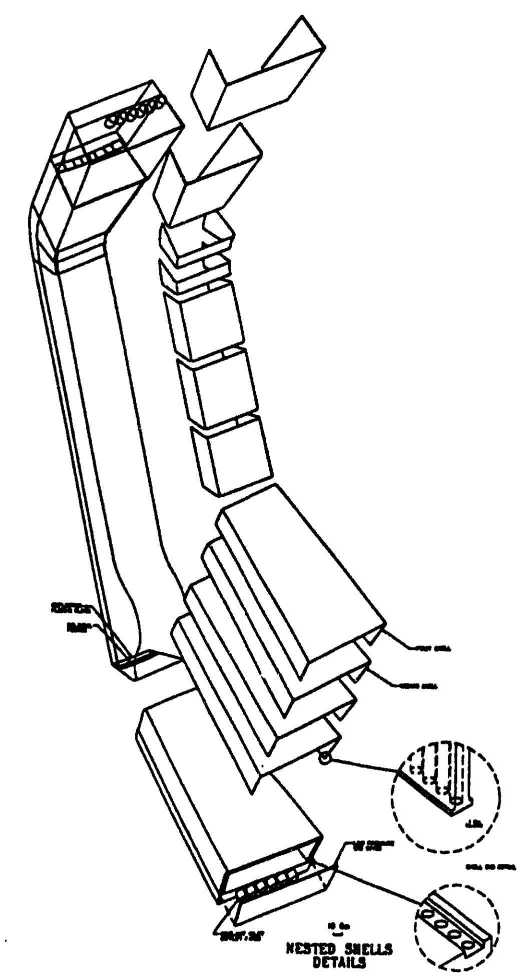

Figure 3

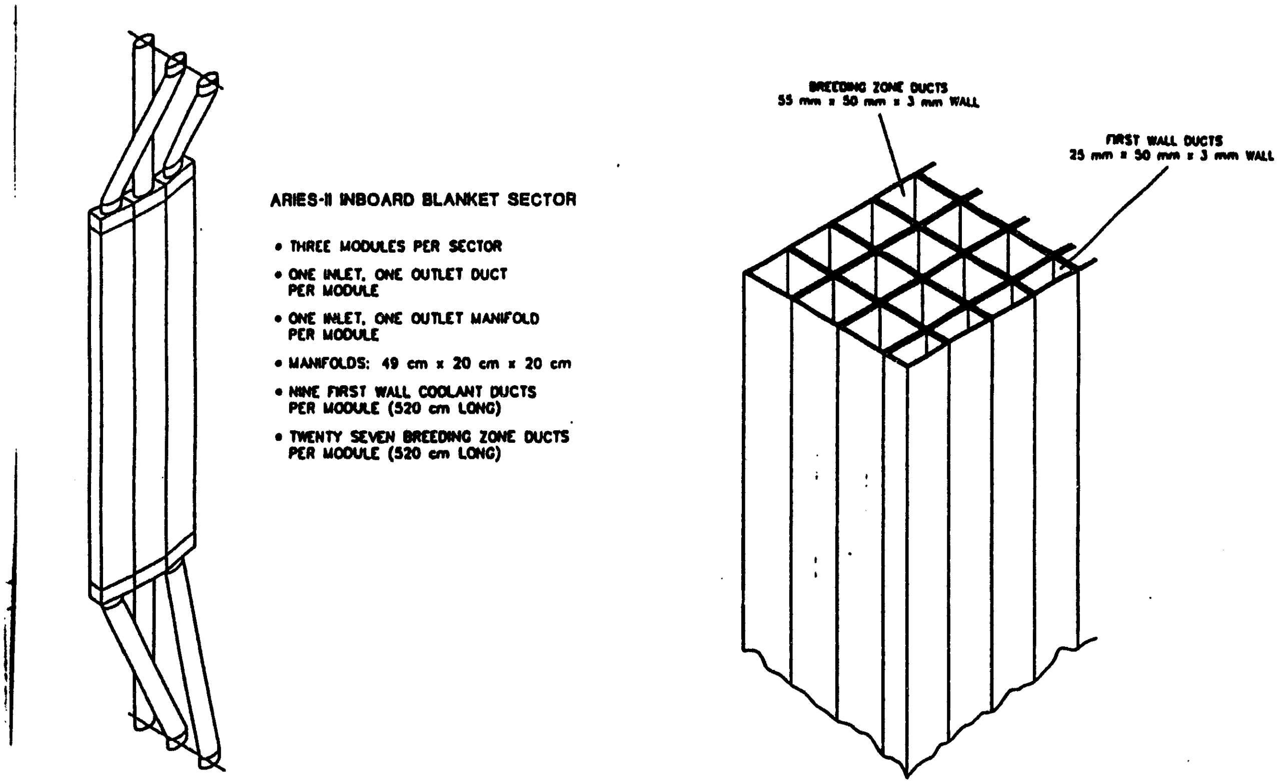

Figure 4

Details of ARIES-III First Wall Design

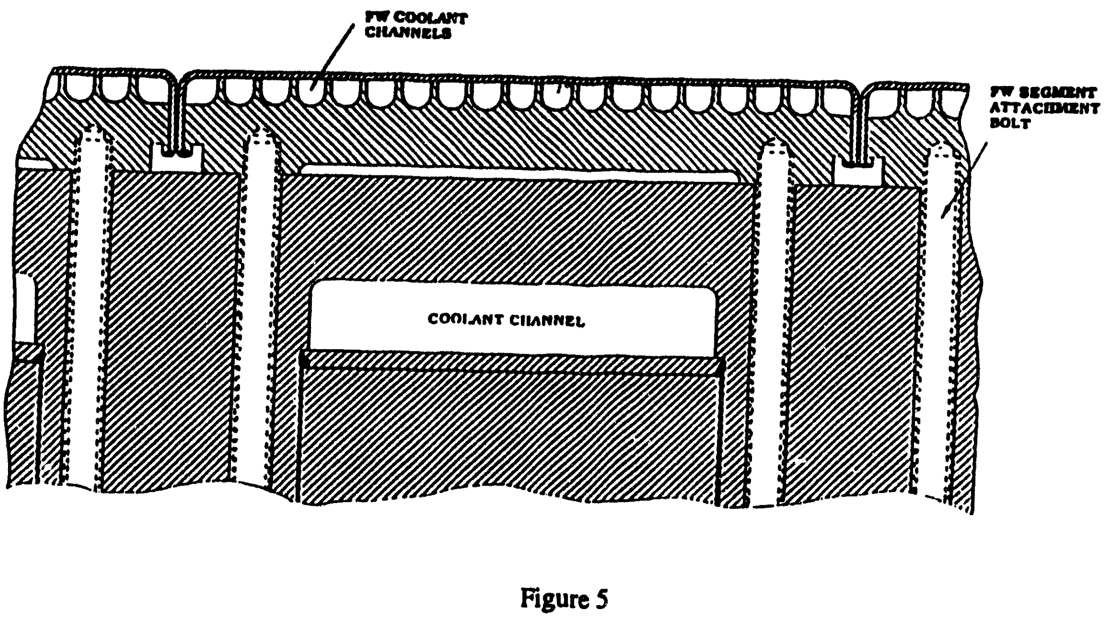

Figure 5

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