Thermo-Mechanical Design and Structural Analysis of the First Wall for ARIES-III, A 1000 MWe D-^{3}He Power Reactor

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Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

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THERMO-MECHANICAL DESIGN AND STRUCTURAL ANALYSIS OF THE FIRST WALL FOR ARIES-III, A 1000 MWe D-^{3}He POWER REACTOR

I.N. Sviatoslavsky, J.P. Blanchard, E.A. Mogahed,
and the ARIES Team
Fusion Technology Institute
University of Wisconsin-Madison
1500 Johnson Drive
Madison, Wisconsin 53706-1687

Abstract

ARIES-III is a conceptual design study of a 1000 MWe D-^{3}He tokamak fusion power reactor in which most of the energy comes from charged particle transport, bremsstrahlung and synchrotron radiation, and only a small fraction ($\sim 4\%$) comes from neutrons. This form of energy is deposited as surface heating on the chamber first wall (FW) and divertor elements, while the neutron energy is deposited as bulk nuclear heating within the shield. Since this reactor does not use tritium, there is no breeding blanket. Instead a shield is provided to protect the magnets from neutrons. The FW is very unique in a D-^{3}He reactor; it must be capable of absorbing the high surface heat in a mode suitable for efficient power cycle conversion, it must be able to reflect synchrotron radiation, and it must be able to withstand high current plasma disruptions. The FW is made of a low activation ferritic steel (MHT-9) and is cooled with an organic coolant (HB-40) at a pressure of 2 MPa. The FW has a coating of 0.01 cm tungsten on the MHT-9, followed by 0.15 cm of Be on the plasma side. This is needed for synchrotron radiation reflection and as a melt layer to guard against the thermal effects of a plasma disruption. Two dimensional finite element models are used with the ANSYS computer code to determine both thermal stresses and temperature profiles. At the midplane, where the surface heating peaks at 1.86 MW/m^2, the bulk coolant temperature is 385°C, the coolant velocity is 16 m/s and the combined primary and secondary stresses are 350 MPa at a bulk steel temperature of 497°C. The maximum Be surface temperature facing the plasma is 571°C.

Introduction

The first wall (FW) in a D-^{3}He fusion power reactor is a very important component of the system because of the multi-functional requirements imposed on it. In the first place, the wall must be capable of reflecting synchrotron radiation. The reason for this is simple; for the synchrotron radiation to deposit its energy in the plasma it must bounce off the confining surfaces many times. If the wall reflectivity is low, the synchrotron radiation gets absorbed in the wall and not in the plasma. For a wall to have a high reflectivity with respect to synchroton radiation it must have a high electrical conductivity. Steel is a poor candidate for that. Copper is excellent, unfortunately it activates and is not environmentally acceptable. Beryllium, however, has a high electrical conductivity, a high thermal conductivity, is neutronically benign with respect to activation and has a vapor pressure which is low enough at its maximum operating temperature (> 600°C) that evaporation is acceptable. For these reasons, it was chosen as the reflecting surface for the FW. A coating of tungsten, 0.01 cm thick, is used between the Be and the steel to prevent diffusion of Be into the steel. The thickness of the Be layer was chosen to be 0.15 cm not from reflectivity considerations but rather from disruption considerations. Aside from the magnetic energy which is released during a plasma disruption, there is also thermal energy deposited on the FW and divertor. The thick coating of Be can mitigate against this occurrence by acting as a sacrificial melt layer, which can be replaced after a plasma disruption.

The FW must also be capable of absorbing the high heat flux coming from charged particle transport, bremsstrahlung and synchrotron radiation in a way that can be efficiently converted in a power cycle. On the order of 96% of the fusion power in a D-^{3}He reactor is absorbed on the surface of the FW and divertor elements. This high surface heat flux must be absorbed as high grade energy while keeping thermal stresses and temperatures at acceptable values.

Finally, the FW must be capable of withstanding the pressures generated when the stored plasma magnetic energy is dissipated during a plasma disruption. When the plasma disrupts, electric currents are induced in the FW and divertor components, and while the currents are decaying, they interact with the toroidal and poloidal magnetic fields inducing forces on the components in which they flow. Fortunately, since a D-^{3}He reactor does not have to breed tritium, the FW and shield can be made very robust. In contrast the FW in a DT reactor is designed very thin in order not to compete with the the breeding material for neutrons. Even though the plasma current in a D-^{3}He reactor is higher than in a DT reactor, the consequences of the magnetic energy dump during a disruption are more manageable.

In this paper we describe the mechanical design, the thermal hydraulics and the structural analysis of the first wall in ARIES-III.

Mechanical Design

There are 20 toroidal field (TF) coils in the ARIES-III reactor. The inboard (IB) shield is divided into 40 equal modules, or two modules per TF coil and the outboard (OB) shield is divided into 80 equal modules, four modules per TF coil. At the midplane, the width of an IB module is 76 cm, allowing for a one centimeter assembly gap between modules. Similarly the OB modules are 84.8 cm wide at the midplane, also allowing for a 1 cm assembly gap. The IB shield is straight from top to bottom and thus the horizontal cross section of the modules remains constant throughout. The OB shield, however, curves following the contour of the plasma. It is widest at midplane where the reactor major radius is 10.93 m and narrowest at the extremities where the radius is 8.0 m.

The first wall is the most important part of the shield. It is designed to be capable of absorbing the incident high surface heat in the coolant while maintaining the thermal stresses at acceptable levels. The structural material in the FW and shield is a modified ferritic stainless steel MHT-9 and the coolant is an organic material designated HB-40. The virtue of the organic coolant is that it can operate at relatively high temperatures while being at a low pressure (at an exit temperature of 425°C the pressure is 2 MPa). Figures 1 and 2 are midplane cross-sections of the OB and IB shields respectively. The first wall is composed of segments divided in the toroidal direction such that the coolant in it flows in the poloidal direction. Figure 3 is a cross section of the FW showing one FW segment flanked by two others. The FW skin is composed primarily of MHT-9, 1.0 mm thick and is coated on the outer surface facing the plasma with tungsten and Be. The W, which is 100 μm thick, acts as a diffusion barrier between the Be and the steel. A coating of Be, 1.5 mm thick is deposited on the W. Its function is to provide a high electrical conductivity at the surface facing the plasma, to reflect synchrotron radiation and act as a sacrificial coating to mitigate against energy deposited by the plasma during a disruption. The FW skin is attached by a continuous spot welding process to a machined MHT-9 steel plate, and the two together constitute the FW segments. Immediately behind the FW skin there is a continuous series of rectangular

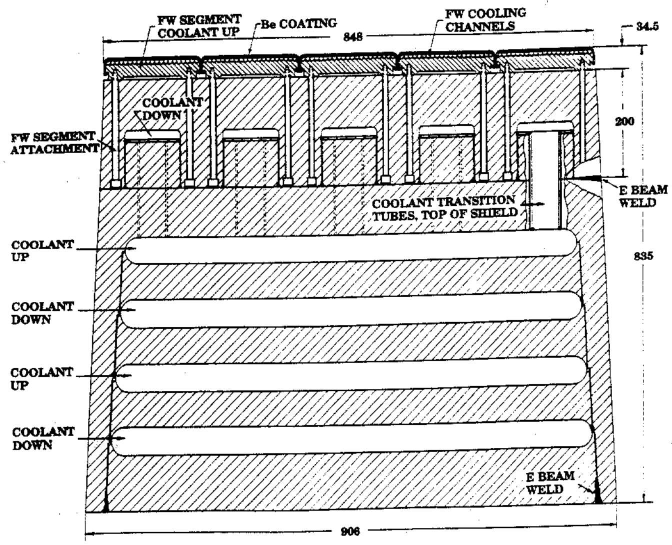

Fig. 1. Midplane cross section of the outboard shield.

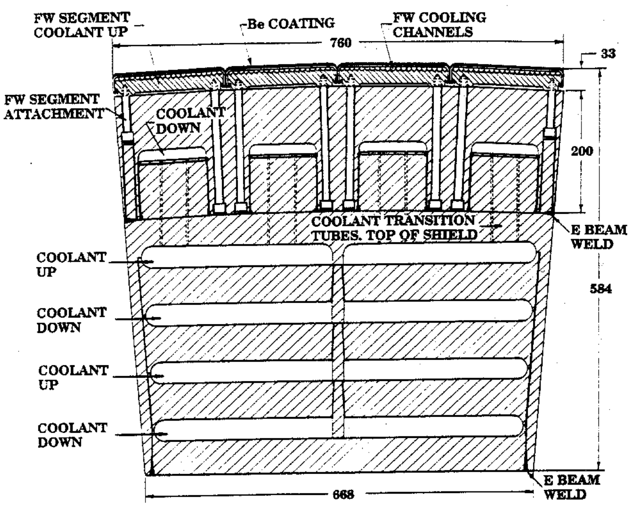

Fig. 2. Midplane cross section of the inboard shield.

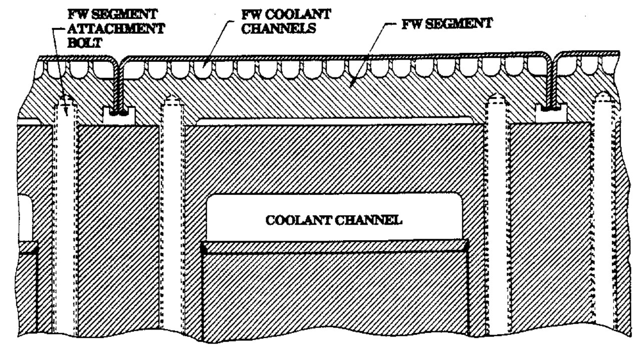

Fig. 3. Cross-section of first wall segments.

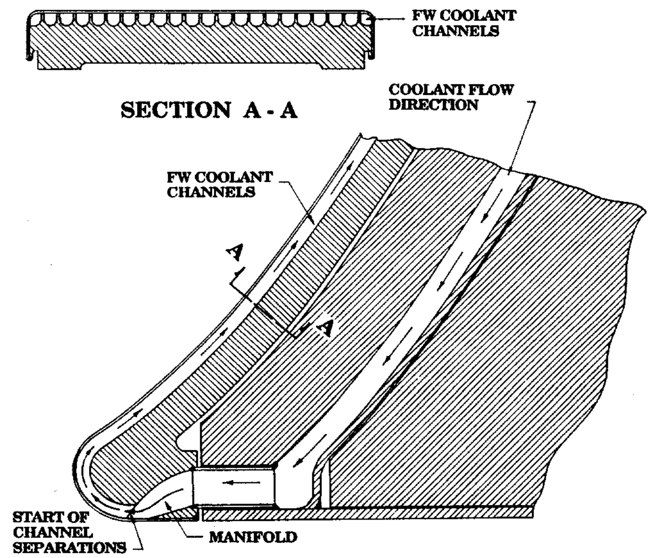

Fig. 4. Coolant transition to FW - lower extremity of OB module.

channels 6 mm wide and 8 mm deep at the deepest point. These channels run in the poloidal direction in a continuous fashion from the bottom to the top. The steel separations between the channels are 1.5 mm thick, and thus the structural fraction in the coolant passage region is 20%. At the interface between the FW segments and the front shield zone, the FW segments are relieved such that they only make contact on the ends where the bolts which attach the segments to the front shield are located. This is done to thermally isolate the FW segments from the shield behind it. There are five FW segments in the OB module each 16.97 cm wide at the midplane and 3.45 cm deep. At the extremities, the two side segments are reduced in width, while the remaining three segments have the same width as at the midplane. The channels in the side segments are shunted to the side of the modules, as the segment width is reduced. At the extremities, the width of the side segments is only 5.46 cm accommodating only 6 of the original 21 channels. Shunting the channels to the side while providing the needed flow area permits a smooth coolant transition with a minimal loss of pressure, and prevents the existence of undercooled areas which will develop into hot spots. The IB FW segments, of which there are 4 per module, are 19 cm wide, and 3.2 cm deep and are of uniform width from top to bottom.

The front part of the shield on which the FW segments are attached is an MHT-9 steel forging 20 cm thick and has a 5% coolant fraction. In the OB shield this piece is curved to conform to the plasma contour and is tapered from the midplane to the extremities. This front shield has the primary function of supporting the FW segments by means of bolts inserted from the back side. A single coolant channel is machined behind each FW segment. This coolant channel cools the front shield and also ducts coolant to the FW segments. Figure 4 shows how coolant is transferred from the front shield to the FW segment. The figure is that of a FW segment which has a constant cross section top to bottom. Obviously, OB end FW segments which are tapered at the extremities will need modification to accommodate the coolant transfer. A similar arrangement is used at the top of the module where the coolant exits from the FW segments and is ducted back to the rear of the shield before it is taken out of the reactor.

Thermal Hydraulics Analysis

Thermal hydraulics calculations for the ARIES-III FW and shield are performed utilizing the output from a one dimensional neutronics model. The surface heat flux on the OB side is $\sim 22\%$ higher than the IB side and thus, we will consider only the OB side in this section.

The FW and shield is cooled with the organic coolant HB-40, which has been successfully used in Canada as a reactor coolant. The major advantage of this coolant over water is that it has a higher boiling point and is capable of higher temperature operation at low pressure. In fact the upper temperature of 425°C is set by thermal decomposition and not by pressure. High temperature operation translates directly into higher thermal conversion efficiency which means lower cost of electricity.

The thermal hydraulics has been carried out with a two dimensional finite element model using the ANSYS [1] program, a commercial com-

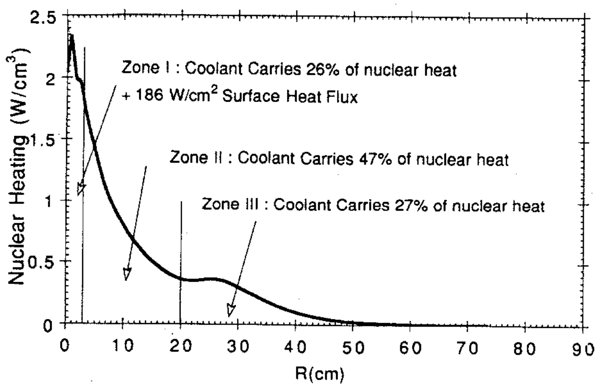

Fig. 5. Nuclear heating profile in OB shield at midplane.

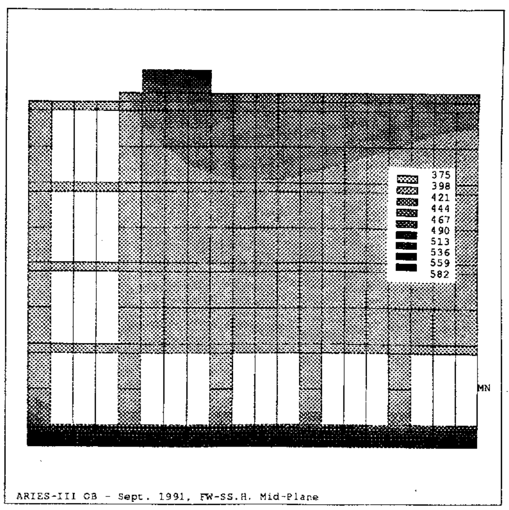

Fig. 6. Temperature contours in a corner of a FW segment.

puter code for calculating thermal stresses and temperature profiles. The following guidelines were used in calculating the thermal hydraulics in the FW:

- Bulk MHT-9 FW structural temperature, 500°C
- Maximum coolant temperature, 425°C
- Maximum surface heat flux, 186 W/cm^3
- Vertical coolant path length in the FW, 8 m.

Figures 1 and 2 indicate the directions of coolant flow in the shields. The coolant makes five vertical passes through the rear shield zones at a low velocity before it reaches the FW in which it flows the entire vertical height from the bottom to the top. The velocity in the FW is 16 m/s and is consistent with that used by the Canadians. This high velocity is needed to achieve a high heat transfer coefficient at the FW to minimize the temperature of the surface facing the plasma. The heat transfer coefficient achieved is 2.19 W/cm^{2}K. Figure 6 shows the nuclear heating distribution in the OB shield. It shows that the coolant absorbs 26% of the nuclear heat and all the surface heat in the FW segments. The remaining 74% of the nuclear heating is picked up in the rear zones of the shield. Recall, however, that nuclear heating accounts for only 4% of the total energy.

Several two dimensional finite element models were created to determine the temperature distribution in the FW and shield. Figure 6 shows the temperature distribution in a corner of a FW segment as represented in an ANSYS model while Fig. 7 shows the same distribution in the middle of the segment, both taken on the OB midplane. The maximum temperature of the front surface of the Be coating is 571°C, while its interface

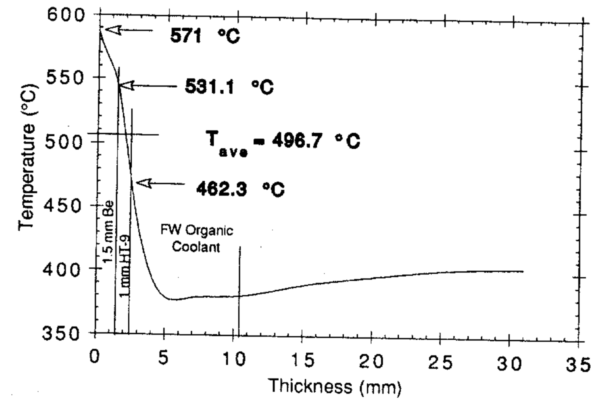

Fig. 7. Temperature profile in the center of a FW segment.

with the steel FW is 531°C. The bulk temperature of the 0.1 cm MHT-9 steel FW is 497°C. Figure 8 shows the temperature distribution extending into the shield right up to the second shield coolant channel. The highest temperature of 565°C occurs on the surface immediately behind the FW segment, which was purposely separated from the FW segment to prevent conduction to the shield and thus degrading the quality of the energy in the FW. This high temperature area is localized and does not compromise the structural integrity of the shield.

First Wall Stress Analysis

The multi-material composite FW is subjected to stresses during all phases of its history, including fabrication, startups, normal operation, shutdowns and restarts. All of these phases have to be considered individually and collectively. Added complications arise from the fact that a plasma disruption can remove a substantial amount of the Be coating. This should in no way affect the structural integrity of the FW. In this section we describe the structural analysis of the FW.

Detailed finite element analyses of the ARIES-III FW have been carried out using the ANSYS [1] code. The model used is shown in Fig. 9 and consists of 887 triangular elements. One side of the model is restrained from motion in the lateral direction, while the other side is allowed lateral motion without rotation at the edge. In other words, the FW is assumed to expand without bending. The loadings considered are steady state surface heating, steady state coolant pressure, steady state volumetric heating, and transient loads caused by the electromagnetic effects of plasma disruptions. Disruption effects manifest themselves as additional pressure distributed uniformly over the FW. Further, the surface heat flux on the OB FW includes a component resulting from radiation from the divertor region. Table I gives the nominal conditions used in the stress analysis.

The ASME code has been used to provide the stress limits in the FW design. At lower temperatures (< 500°C) where thermal creep is not important, the design limit is taken as 1/3 of the ultimate strength or 2/3 of the yield strength. These design stresses are shown in Fig. 10 for HT-9 steel in the temperature range relevant to this study. It shows that 1/3 of the ultimate strength curve is lower than the 2/3 of yield up to a temperature of 530°C and, hence we have opted to use it. The design limit at 500°C is 150 MPa. Above 500°C [2], creep becomes

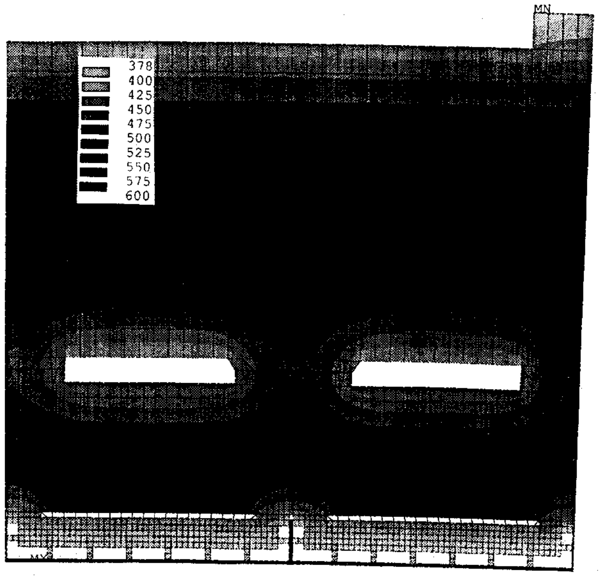

Fig. 8. Temperature contours in front part of OB shield.

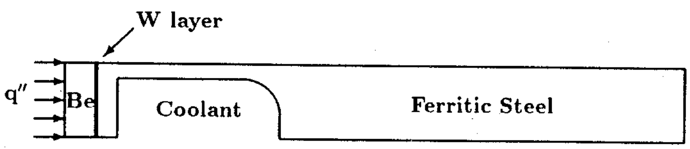

Fig. 9. First wall model used in the structural analysis.

significant and for this reason we have decided to limit the FW bulk steel temperature to $< 500°C$.

Four different stages of loading are considered in the structural analysis: fabrication, startup, steady state operation and plasma disruptions. Residual stresses due to fabrication are very low. The hot forming of the steel FW segments will induce stresses which will be annealed out before it is welded to the steel backing plate, and the stresses induced by the spot welding will be low. Startup in the ARIES-III reactor will be with an initial phase of DT which will provide 80% in bulk nuclear heating allowing the FW temperature to rise uniformly. This uniform temperature rise to 385°C will induce stresses of ~ 10 MPa in the FW.

The primary stresses induced during normal operation are caused by the coolant pressure. Here we do not take credit for the FW thickness due to the Be coating. The peak primary stress in the bare FW is 59 MPa. The ASME code limits the combined primary and secondary stresses not to exceed three times the allowable which in this case would be 450 MPa. During normal steady state operation we find that the peak combined primary and secondary stresses are 350 MPa, well below the allowable. Because a disruption may remove a substantial fraction of the Be coating, it should be verified that even an uncoated FW will be within design limits. The maximum combined stresses in a bare wall are 397 MPa, also well below the allowable.

The main areas of concern due to a plasma disruption are: a significant fraction of the Be coating is lost; the structural temperature will rise due to Joule heating from the induced currents; and the electromagnetic forces induced. We have already shown that structurally, an uncoated wall poses no problem. We have estimated the temperature rise due to Joule heating to about 50°C, or the bulk steel temperature to 550°C. The integrated increase in the creep strain of the FW is minimal as long as the number of disruptions over the lifetime of the reactor is kept low. This number would have to be in the tens of thousands before it becomes a

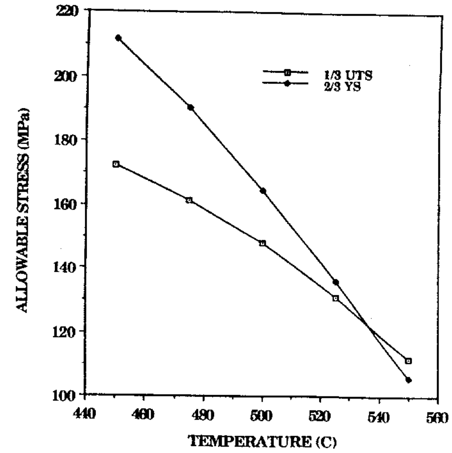

Fig. 10. Allowable stresses in HT-9 structure.

Table I. Nominal Parameters Used in the Stress Analysis

Surface heating (MW/m^2)	1.86
Vol. heating in the steel (W/cm^3)	2.18
Vol. heating in the Be (W/cm^3)	1.90
Heat tran. coeff. at FW (W/cm^2K)	2.19
Bulk coolant temp. (C)	385
Coolant pressure (MPa)	2.0
Disruption pressure (MPa)	0.13

problem. Finally, the induced pressure due to a disruption is ~ 0.13 MPa uniformly distributed across the FW. This is small compared to the 2 MPa coolant pressure and will raise the normal stresses from 59 MPa to 63 MPa.

Summary and Conclusions

We have shown that the first wall in the ARIES-III D-^{3}He power reactor, made of MHT-9 and cooled with the organic coolant HB-40 can be relatively simple in design and can be made to meet the synchrotron reflectivity requirements and conform to all the design limitations both thermal and structural. The most severe limitation is on the bulk temperature of the steel FW (500°C) which restricts it to a thickness of 0.1 cm. A major inconvenience in the event of a disruption is the need to recoat the Be reflecting surface. It is hoped that by the time such reactors will be in commercial operation, disruptions will be well understood and their frequency kept very low.

Acknowledgement

We acknowledge the contributions of D. K. Sze to this design, who proposed and took the lead in implementing the use of an organic coolant for ARIES-III. This reasearch has been supported by the U.S. Department of Energy.

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