# ARIES-CS Loss of Coolant and Loss of Flow Accident Analysis

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# ARIES-CS LOSS OF COOLANT AND LOSS OF FLOW ACCIDENT ANALYSES

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*Loss of Coolant Accident (LOCA) and Loss of Flow Accident (LOFA) thermal simulations have been performed for the ARIES compact stellarator fusion power plant. The ARIES-CS design uses three separate coolant loops: lithium-lead (LiPb) in the blanket, helium in the blanket and the shield, and water in the vacuum vessel. The thermal response to LOCA/LOFA conditions was simulated using transient axisymmetric finite element models. In these analyses, the plasma was quenched three seconds after coolant loss, and the temperature of the chamber components subsequently increased due to the generated decay heat. Thermal simulations determined the maximum temperatures reached in the various components were below the 740$^o$C temperature limit for the reusability of the ferritic steel structure.*

## I. INTRODUCTION

The thermal consequences of Loss of Coolant Accident (LOCA) and Loss of Flow Accident (LOFA) events have been analytically investigated for the ARIES-CS fusion power plant design. The ARIES-CS is a 1000 MW fusion power plant conceptual design launched to study the physics and engineering characteristics of compact stellarator (CS) power plants.[1] These analyses predict the transient temperatures of the various chamber components in the event of such accidents, and their impact on the reusability of the structural components. The ability to withstand a LOCA/LOFA event without damage to the power core is critical, and similar analyses were performed for previous ARIES configurations.[2-5] After loss of coolant or coolant flow, temperatures slowly rise due to the generated decay heat, and while the probability of such an accident is low, it is important that the structural temperatures stay below the 740$^o$C reusability temperature limit of ferritic steel (FS), used in each component. Finite element analysis was used to simulate the transient thermal response of the ARIES-CS design to LOCA/LOFA scenarios, and the maximum temperatures in each component were tracked for the 30 days of the simulation.

The ARIES-CS design uses three coolant types: lithium-lead and helium in the blanket, helium in the shield, and water in the vacuum vessel. This creates the possibility for a number of LOCA/LOFA combinations, but this paper will focus on the most likely condition of loss of flow in the LiPb and water and loss of coolant in the helium. Results for an extreme condition with loss of LiPb in addition to the helium will also be briefly discussed. The limited space for the internals and the varying cross-section of the compact stellarator makes a single wall radial build impractical.[2] The zero degree cross-section of the power core, Fig. 1, illustrates how the clearance between the plasma channel and coil support tube varies around the perimeter. The full blanket design, illustrated in the top of Fig. 2, covering 61% of the total first wall area is used where space permits (most of the right hand side of the 0$^o$ toroidal angle cross-section in Fig. 1). Another 24%, due to space limitations, will have a reduced thickness design called the non-uniform blanket (the left side in Fig. 1). The minimum thickness version of the non-uniform blanket is illustrated in the bottom half of Fig. 2. Both radial builds have been analyzed under LOCA/LOFA conditions. The remaining surface area (15%), covered by divertors (top and bottom of plasma channel in Fig. 1), is not considered in this paper.

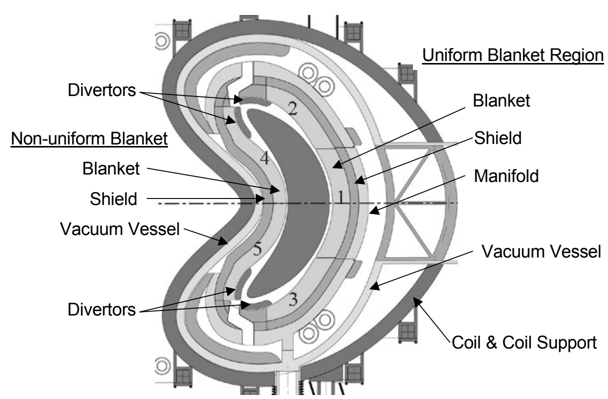


Fig. 1 Cross-section of power core at 0$^o$ toroidal angle showing full and non-uniform blanket regions and divertors.

## II. SYSTEM AND ANALYSIS DESCRIPTION

Radial builds for the full blanket and non-uniform blanket regions are illustrated in Fig. 2. The first wall and blanket use a dual coolant option where helium (He) cools the FS structure while a lithium-lead (LiPb) mixture flows slowly through the breeding zone. Further details of the coolant flows and materials in the full and non-uniform blanket configurations may be seen Fig. 3. A 0.5 cm thick SiC liner helps maintain the FS temperature below 600°C during operation. The shield is cooled by helium flowing in FS channels around a borated steel filler. Behind the shield lies the coolant manifolds that contain the plumbing for the LiPb and helium coolant loops. The vacuum vessel, separated from the manifold by at least a 2 cm gap, is cooled with water and operates at a much lower temperature than the blanket and shield. Outside the vacuum vessel is the winding pack separated from the VV by a gap and thermal insulation. Because the insulation must be highly efficient to maintain cryogenic temperatures in the magnets, the outer surface of the vacuum vessel is treated as adiabatic in these analyses.

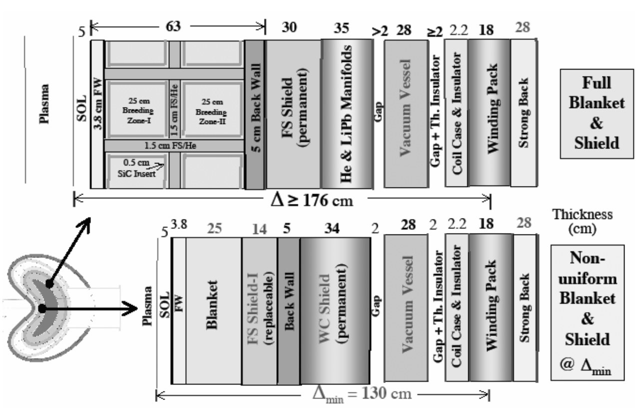


Fig. 2. Schematic of the ARIES-CS radial builds for the full blanket and non-uniform blanket regions.

For the non-uniform blanket, the replaceable blanket and FS shield are similar to those in the full blanket region but with reduced total thickness. Outside the FS shield is a tungsten carbide (WC) permanent shield unique to the non-uniform blanket regions. The permanent shield is also cooled with He. There are no manifolds in the non-uniform blanket regions, and the vacuum vessel design is the same as the full blanket.

The presence of three coolants makes possible a variety of LOCA and LOFA events. Because of the high reliability of the water coolant system in the vacuum vessel, only a loss of flow in the VV is considered with natural convection occurring between the water and the VV. The water is assumed to flow naturally carrying heat out of the power core. For the helium in the blanket and shield, only LOCA conditions are considered. In a He LOFA event, the presence of non-flowing gas would have minimal thermal effect and results would be very similar to LOCA. For the LiPb in the blanket, the analyses described here focus on the LOFA condition because of the long time frame (hours) required for the LiPb to drain from the system, and the LiPb was assumed to be non-flowing with afterheat. It is also assumed that the loss of coolant or flow occurs in all 200 modules. The modular nature of the ARIES-CS design would make this scenario less likely, and some of the heat from a failed module could be removed through the adjacent cooled modules, but the system wide LOCA/LOFA events represent a potential worst-case scenario.

There are a variety of assumptions required for these analyses that are important to detail. The finite element

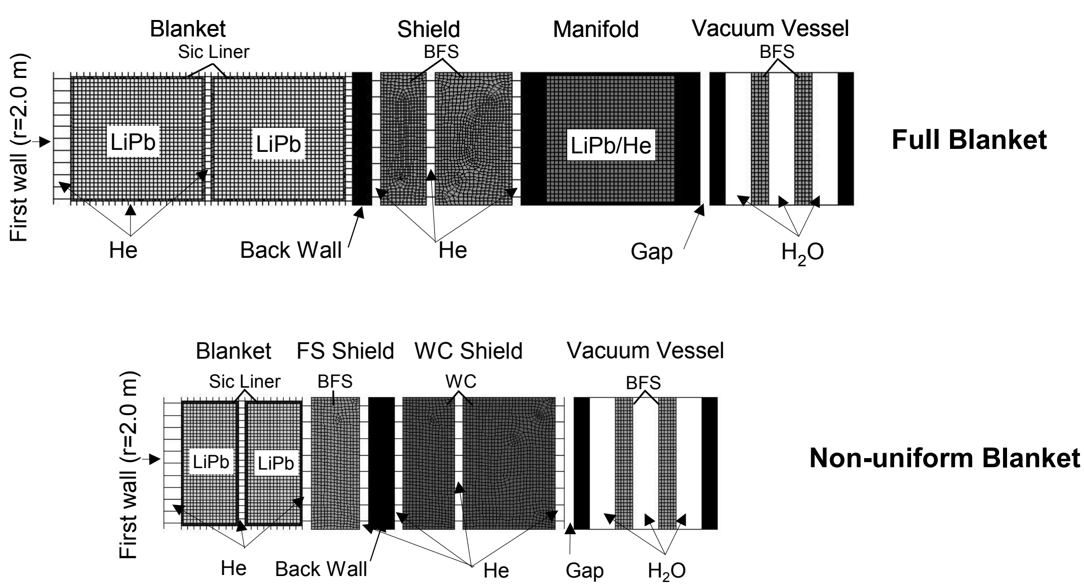


Fig. 3. Finite element models for the full and non-uniform blankets showing materials and coolants.

modeling and analysis was performed using the ANSYS code. The models used axisymmetric elements to model the details of components. The FE models used for the full and non-uniform blanket analyses are illustrated in Fig. 3. Because of the repeating nature of the build geometry, the top and bottom edges of the model may be treated with symmetry boundary conditions, and the first wall surface is treated as adiabatic because the facing surfaces in the plasma channel will be at a similar temperature. While these models do not capture the toroidal nature and varying cross-section of the complex compact stellarator geometry, they do provide a good estimate of overall system performance at reasonable modeling and computational costs. The modeling also assumes that the outside of the vacuum vessel is adiabatic, and heat is only removed from the system by natural convection to the water in the vacuum vessel. This convection occurs with a coefficient of 500 W/m$^2$-C, and the water remains at 140$^o$C. Also, in the event of an accident, the plasma remains on for three seconds. In addition to conduction, radiation heat transfer is modeled in the empty coolant passages and the gap between the manifold and vacuum vessel. An emissivity of 0.3 is used for all radiating surfaces.

Decay heats were calculated for 20 discrete heating zones in the various components. These heats were applied as time dependent volumetric heating at 18 time points. Examples of the decay heat in various full blanket components are illustrated in Fig. 4. The first wall has the highest heating. The average first wall decay heat is 730 kW/m$^3$ immediately after shutdown of the plasma, but drops an order of magnitude to 70 kW/m$^3$ within one day. Heating falls off quickly through the depth of the blanket with the back wall having a maximum decay heat of only 4 kW/m$^3$. For the non-uniform blanket, first wall heats are similar to those of the full blanket, but other decay heats are generally higher largely because of the reduced breeding area of the non-uniform blanket and the presence of the WC filler in the shield. As mentioned previously, plasma energy remains on for 3 s following loss of coolant. During this time, predicted fusion heating values (ranging in the blanket from 22 MW/m$^3$ at the first wall to 0.4 MW/m$^3$ at the back wall) are applied to the various components. Additionally, a surface heat of 0.76 MW/m$^2$ is applied to the first wall to simulate heating from the plasma during the 3 s. Initial temperatures at the onset of LOCA/LOFA are applied to the various structural components and coolants based on the maximum operational temperatures expected. These initial values are summarized in Table I.

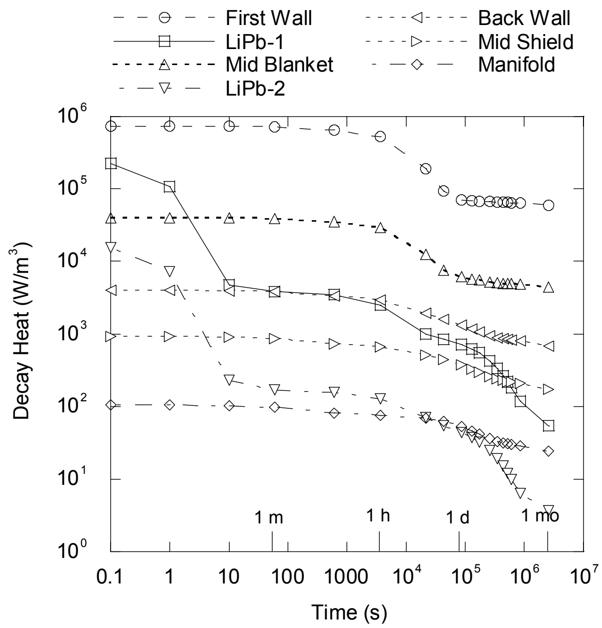

Fig. 4. Decay heats in various components of the full blanket radial build.

TABLE I. Temperatures at Onset of LOCA

| Component | Initial Temperature $^o$C |
|---|---|
| Front Wall | 571 |
| Blanket FS | 500 |
| Blanket SiC | 530 |
| Blanket LiPb | 617 (average) |
| Back Wall FS | 450 |
| Shield FS | 450 |
| Manifolds FS | 450 |
| Manifolds LiPb | 617 (average) |
| VV Water | 140 |
| VV FS | 180 |

## III. RESULTS AND DISCUSSION

The finite element models, heat loads, and conditions previously described were used to calculate the transient temperature response from the accident onset until the analysis termination time of 30 days. Results for the full blanket design are plotted in Fig. 5 for LOCA in the He and LOFA in the LiPb and water. The maximum temperature, 725$^o$C, occurs in the first wall immediately after shutdown of the plasma (3 s). This indicates the critical factor for LOCA/LOFA first wall temperature is the time between loss of coolant and the plasma quench as the first wall rises from 571$^o$C to 725$^o$C in the 3 s assumed in this analysis. After plasma shutdown, first wall temperatures fall briefly and then begin to rise again. The other components remain significantly cooler; the maximum temperatures of the back wall (545$^o$C) and shield (524°C) are significantly below the 740°C reuse temperature of the ferritic steel. Temperatures in the vacuum vessel remain below 180°C and approach the 140°C water temperature with increasing time. The maximum temperatures in all the components occur within the first 12 hours after shutdown, and all temperatures fall continuously after 12 h indicating that longer term LOCA/LOFA conditions would not be a problem as long as natural convection in the water loop is maintained.

Because the maximum front wall temperature occurs immediately after the shutdown of the plasma, it is highly dependent on the heat inputs during this time. Heat is applied to the first wall as volumetric nuclear heating and surface heating from the plasma, with the surface heating representing 90% of the total input. A value of 0.76 $MW/m^2$ was applied in the analyses, and the response is very sensitive to this value. When the surface flux is reduced to 0.5 $MW/m^2$ the first wall temperature after 3 s of fusion power is 676°C and the overall maximum temperature is 687°C. So more attention will need to be paid in the future to the assumptions about time to plasma quench of the fusion heat input during this time as they are critical to the LOCA/LOFA temperature response.

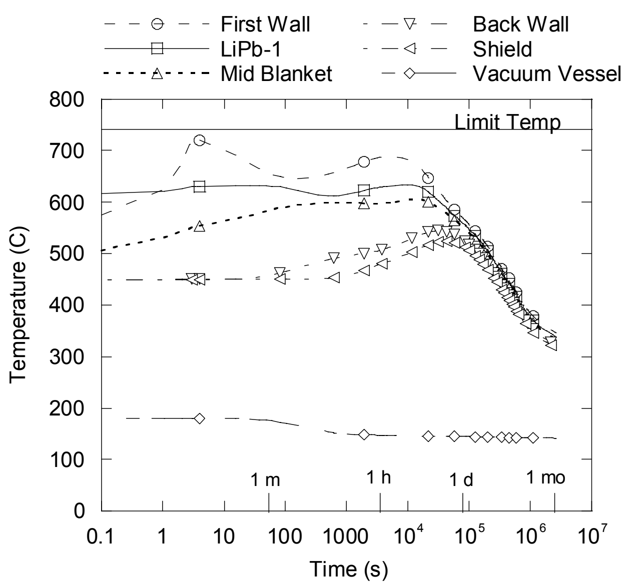


Fig. 5. Transient temperatures from LOCA/LOFA of various components of the full blanket.

For the non-uniform blanket regions, the maximum temperature occurs again in the first wall immediately after shutdown of the plasma at 3 s, as illustrated in Fig. 6. The maximum temperature in this case of 730°C is higher than that of the uniform blanket. This is most likely due to the higher average nuclear heating of the reduced thickness LiPb layer. After this peak, first wall temperatures fall and then begin to rise with the rest of the component temperatures. The second first wall peak reaches 701°C, and temperatures reach 678°C in the FS shield and 661°C in the WC shield. These maxima occur approximately one day after loss of coolant, and then all component temperatures fall continuously. The differences in the behaviors of the two blanket regions can be attributed to the significantly higher decay heats of the WC material in the non-uniform blanket.

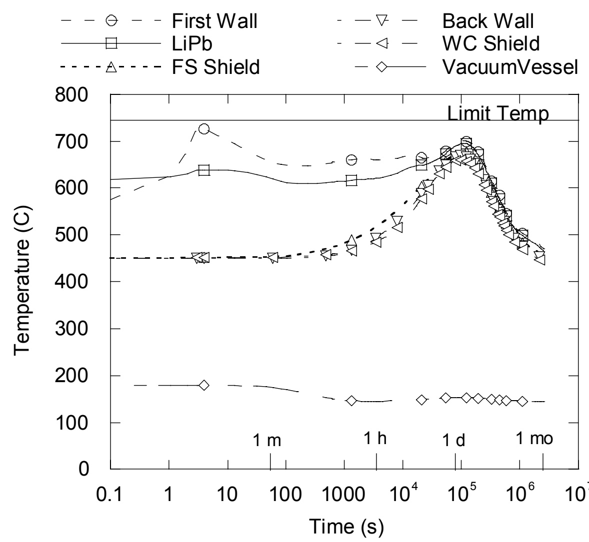


Fig. 6. Transient temperatures from LOCA/LOFA of various components in the non-uniform blanket.

A scenario where the LiPb coolant in the blanket is lost in addition to the helium was also analyzed (LOCA in the He and LiPb, LOFA in the water). Because of the long drainage time (several hours) for the LiPb in the event of some type of failure, this event is unlikely to occur and is considered a beyond design basis accident. For these analyses, the LiPb in the blanket and manifold was removed, and radiation across the empty coolant channels was added. The SiC surfaces inside the LiPb passages were assumed to have an emissivity of 0.5, below the expected values of 0.8 to 0.9 to account for potential reductions from residual LiPb. The decay heats and initial temperatures remained the same, but the 3 s of nuclear heating was modified to reflect the removal of the LiPb. Once again the peak temperatures occurred immediately after the 3 s of nuclear heating following the onset of LOCA/LOFA; the peak first wall temperatures of 725°C for the full blanket and 730°C for the non-uniform were unchanged. The second peak in temperature produced by the decay heats is slightly higher with the loss of the LiPb. It was also found that the maximum temperature was quite sensitive to the emissivity used in the SiC in the LiPb channels. For instance, reducing the emissivity to 0.3 increased the maximum first wall temperature to 753$^o$C. For the full blanket, the first wall reaches 697$^o$C, and for the non-uniform blanket the second maximum is 722$^o$C. Once again the peak temperatures are below the 740$^o$C reuse temperature of the ferritic steel, and the unlikely rapid loss of the LiPb should not necessitate component replacement.

## IV. CONCLUSIONS

Loss of coolant and loss of flow accident analyses were performed for the ARIES-CS fusion power plant conceptual design. These transient finite element analyses simulated LOCA/LOFA scenarios for 30 days following an event for both the full blanket and non-uniform blanket radial builds. Maximum temperatures for both radial builds occurred in the first wall after the 3 s of nuclear heating following loss of coolant. The temperatures approached the 740$^o$C reuse temperature limit for ferritic steel. Maximum temperatures are 725$^o$C for the full blanket and 730$^o$C the non-uniform blanket build. These results indicate the critical factor for reusability of the first wall and blanket is the timely shutdown of the plasma following coolant loss. Also noted were second peaks in first wall temperatures that were produced solely by the decay heat in the components. For the case of LOCA in the He and LOFA in the LiPb and water, this maximum was 688$^o$C for the full blanket and 701$^o$C for the non-uniform blanket build. An additional, off-design condition of LOCA in both the He and LiPb was also analyzed and briefly discussed. For this case, maximum first wall temperatures, which occurred immediately after shutdown of the plasma, were also below the component reuse temperature. Finally, it should be noted that these results are predicated on heat removal from the system through natural convection to the water in the vacuum vessel as the remainder of the system is well insulated by the magnet insulator. Loss of the VV heat removal mechanism would result in the rapid overheating of the system.

## ACKNOWLEDGMENTS

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## REFERENCES

[1] A. R. RAFFRAY, L. EL-GUEBALY, S. MALANG, X. WANG, the ARIES Team, "Attractive Design Approaches for a Compact Stellarator Power Plant," *Fusion Science and Technology*, **47**, 422 (2005).

[2] R.L. MILLER and the SPPS Team, "Stellarator Power Plant Study," University of California San Diego Report UCSD-ENG-004 (1997).

[3] E. MOGAHED et al., "ARIES-RS Loss of Coolant Accident (LOCA) Analysis," Proceedings of IEEE 17th Symposium on Fusion Engineering, San Diego, CA, 1, 204-207 (1997).

[4] E. A. MOGAHED, "Loss of Coolant Accident (LOCA) Analysis of the ARIES-ST Design," *Fusion Technology*, **34**, pp. 1079-1083 (1998).

[5] E. MOGAHED, L. EL-GUEBALY, A. ABDOU, P. WILSON, and D. HENDERSON, "Loss of Coolant and Loss of Flow Accident Analyses for ARIES-AT Power Plant," *Fusion Technology*, **39**, No. 2, 462 (2001).

[6] L. EL-GUEBALY, P. WILSON et al., "Designing ARIES-CS Compact Radial Build and Nuclear System: Neutronics, Shielding, and Activation," to be published in *Fusion Science and Technolog.*