# ARIES-ACT-1 Loss of Coolant/Flow Accident Thermal Analysis

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October 2013

UWFDM-1416

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***Abstract***

*Thermal simulations of a loss of coolant and flow accidents have been performed for the ARIE-ACT-1 fusion power plant. The design uses three separate coolant loops: lithium-lead (LiPb) in the blanket, helium in the divertor, structural ring, and vacuum vessel, and water in the low temperature shield. The thermal response to total loss of helium and water and loss of LiPb flow was simulated using transient, axisymmetric finite element models. In these analyses, the plasma was quenched three seconds after the onset of 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 above the 750$^o$C temperature limit for the reusability of several ferritic steel-based components, but lower than the 1050$^o$C temperature limit for advanced nano-structured ferritic alloys.*

## I. INTRODUCTION

The thermal consequences of Loss of Coolant Accident (LOCA) and Loss of Flow Accident (LOFA) have been analytically investigated for the ARIES-ACT-1 fusion power plant design[1]. This tokamak design features aggressive physics and technology, using a silicon carbide (SiC) blanket concept with LiPb as coolant and breeder. The ARIES-ACT-1 design employs three coolant types: lithium-lead in the blanket, helium in the divertor, structural ring and vacuum vessel (VV), and water in the low temperature (LT) shield. This creates the possibility for a number of LOCA/LOFA combinations. Loss of flow accidents have already been considered[2] so this paper will focus on the extreme condition of loss of flow in the LiPb and loss of coolant in the helium and water which represents a worst case, less probable, accident scenario. The configuration of the tokamak power core (Fig. 1) complicates the removal of heat when coolant is lost. A layer of highly efficient superinsulation surrounds the LT shield to maintain the 4 K cryogenic temperature required by the superconducting coils. This leaves the outboard maintenance ports, shown in more detail in Fig. 2, as the primary paths for heat removal from the power core, hence heat removal from the inboard sections is problematic. In the event of loss of coolant, decay heat must be transferred from the inboard to the outboard regions by conduction through the structural ring, VV, and LT shield, radiation across the plasma chamber, and then out of the vessel by conduction through the maintenance ports and radiation to the cryostat. These analyses predict the transient temperatures of the various chamber components in the event of such an accident, 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.[3-6]
After loss of coolant or 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 projected 750$^o$C reusability temperature limit of ferritic steel (FS)[7,8] used in the structural ring, vacuum vessel and LT shield. A transient temperature exceeding 750$^o$C suggests the use of the more advanced nano-structured ferritic alloys (NFA) that can be reused if the temperature reaches 1050$^o$C during an accident[7,8]. Finite element (FE) analysis was used to simulate the transient thermal response of the ARIES-ACT-1 design to the LOCA/LOFA scenario previously described, and the temperatures of each component were tracked for one year after the onset of accident. Additionally, the FE model is used to evaluate the effects of various modeling assumptions and a potential change to the inboard radial build on post accident temperatures.

## II. SYSTEM AND ANALYSIS DESCRIPTION

Radial builds for the outboard and the original inboard regions are illustrated in Figs. 3 and 4 along with the material compositions for each component[9]. The blankets use the SiC/SiC composite structure while the lithium-lead (LiPb) eutectic flows through the breeding zone. The average operating temperature of the blankets is about 850$^o$C. The structural steel ring behind the blanket is cooled by helium flowing in channels and operates at approximately

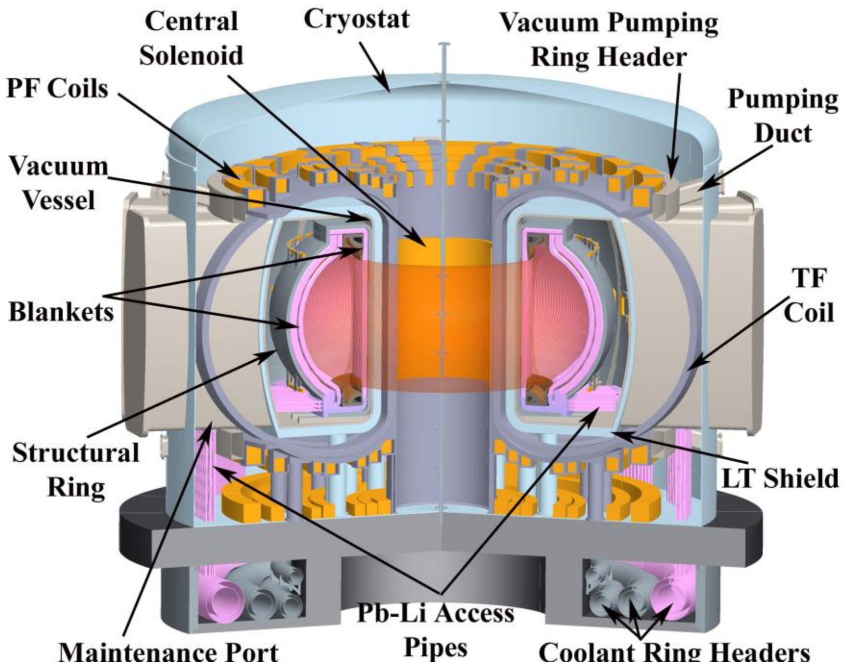


Fig. 1. Schematic of the ARIES-ACT-1 power core.

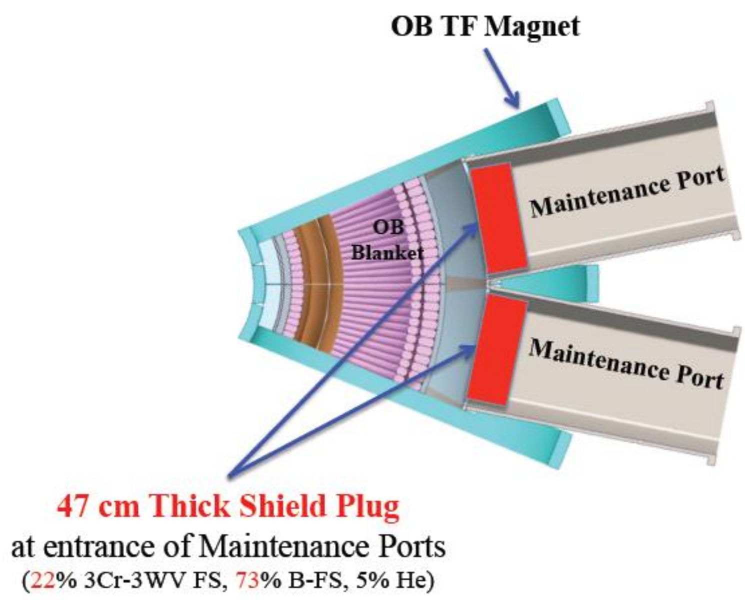


Fig. 2. Schematic of the maintenance port region and shield plugs.

675$^o$C. Behind the structural ring lies the VV which is also helium cooled but operates at a substantially lower temperature (300$^o$C). The LT shield is cooled with water and operates at a much lower temperature (25$^o$C) than the blanket and structural ring. The outboard LT shield consists of a FS structure surrounding a borated steel (B-FS) filler. The inboard shield is constructed from FS and tungsten carbide (WC) filler. A modified inboard radial build, with the WC LT shield replaced by a slightly thicker B-FS shield, is shown in Fig. 5. Replacing the WC with a B-FS filler reduces the decay heat in the inboard LT shield which should reduce structural temperatures in the event of the accident described in this paper. Outside the LT shield is the magnet separated from the shield 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 LT shield is treated as adiabatic in these analyses. On the outboard side, much of the external surface is covered by maintenance ports as illustrated in Figs. 1 and 2. The port regions do not have a VV and LT shield, but rather a 47 cm thick He-cooled shield plug made from FS and B-FS as shown in Fig. 2. The shield plugs operate at 300$^o$ C and are not covered by superinsulation, and thus heat may be removed from the power core by conduction in the port walls and radiation from the shield plug to the port door and eventually to the surface of the cryostat which is exposed to the ambient atmosphere.

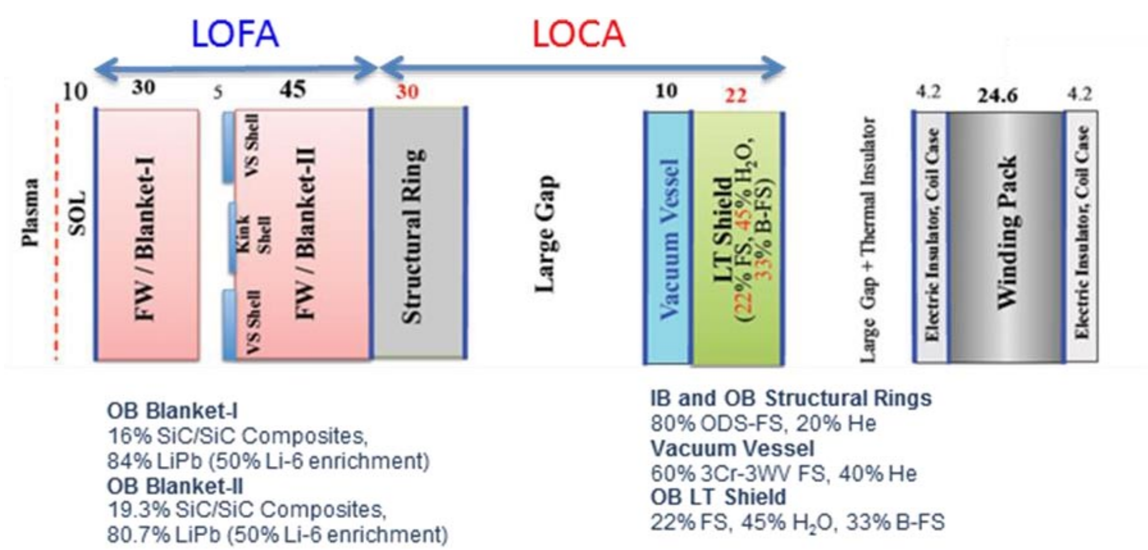


Fig. 3. Schematic of the outboard radial build with component material compositions.

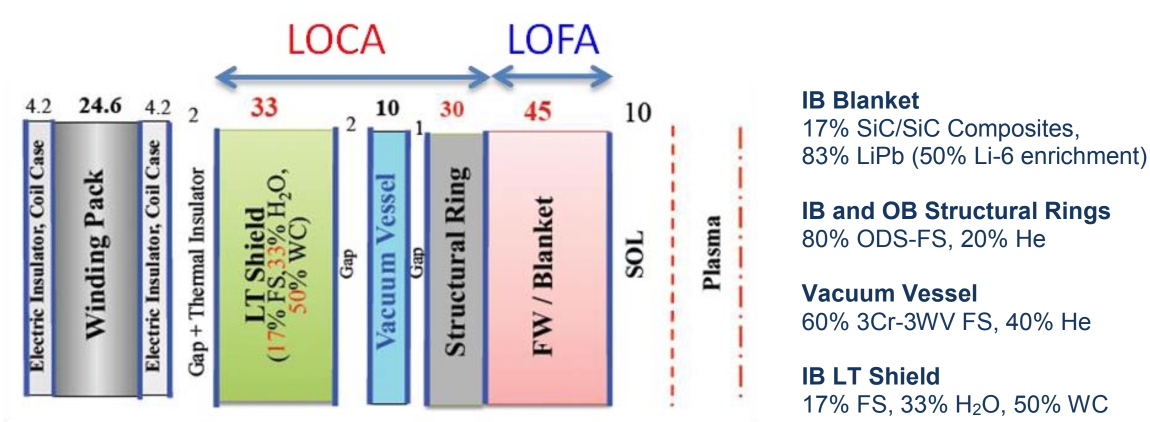


Fig. 4. Schematic of the inboard radial build with component material compositions with the tungsten carbide filler in low temperature shield.

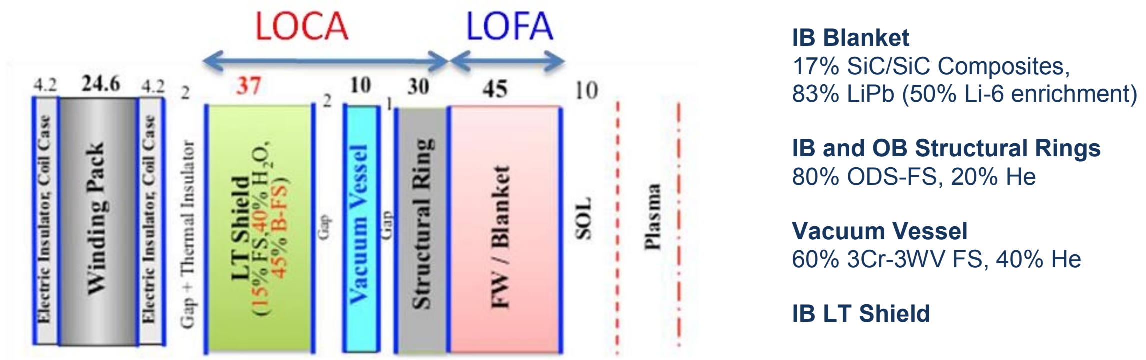


Fig. 5. Schematic of the inboard radial build with component material compositions with the borated ferritic steel filler in low temperature shield.

These preliminary thermal models required a number of modeling simplifications and analysis assumptions in order to meet both time requirements and accommodate an evolving design. To begin, the complex, three-dimensional system was approximated with a 2-D axisymmetric finite element model which ignores potential local heat transfer paths such as ducts, piping, and supports, but models the power core and conduction paths for heat removal reasonably well. Material properties for the various components were computed using a simple rule of mixtures. For the maintenance port regions, conduction and radiation areas were scaled to allow the axisymmetric model to represent the 3-D structure. Similarly, a horizontal symmetry plane was assumed, and the geometry simplified to facilitate rapid modeling. The finite element model used is shown in Fig. 6. Radiation across the gaps in the radial builds was simplified so that elements only radiated to elements directly across from them, an assumption that may not be entirely valid for the larger gaps but avoided complicated view factor calculations. Initially, it was assumed that radiation across the plasma channel would be negligible and could be ignored, but this assumption was found to be invalid as maximum inboard first wall (FW) temperatures (1730$^{o}$C) were much higher than outboard FW temperatures (1000$^{o}$C) indicating significant radiation would occur. Radiation across the plasma channel was modeled using two-dimensional axisymmetric view factor calculations for the first wall and divertor surfaces. The emissivity of the silicon carbide surfaces was assumed to be 0.9 which is in the middle of reported values.[10] For metallic surfaces, the emissivity is highly dependent on surface finish and oxidation, and an initial value of 0.2 was assumed. At the external boundary, the temperature of the cryostat was fixed at 30$^{o}$C. The simplified representations of the divertors and their supporting structure have been included in these models so that their heat loads are included in the total system, but the level of detail in the modeling precludes accurate temperature predictions for the divertor components.

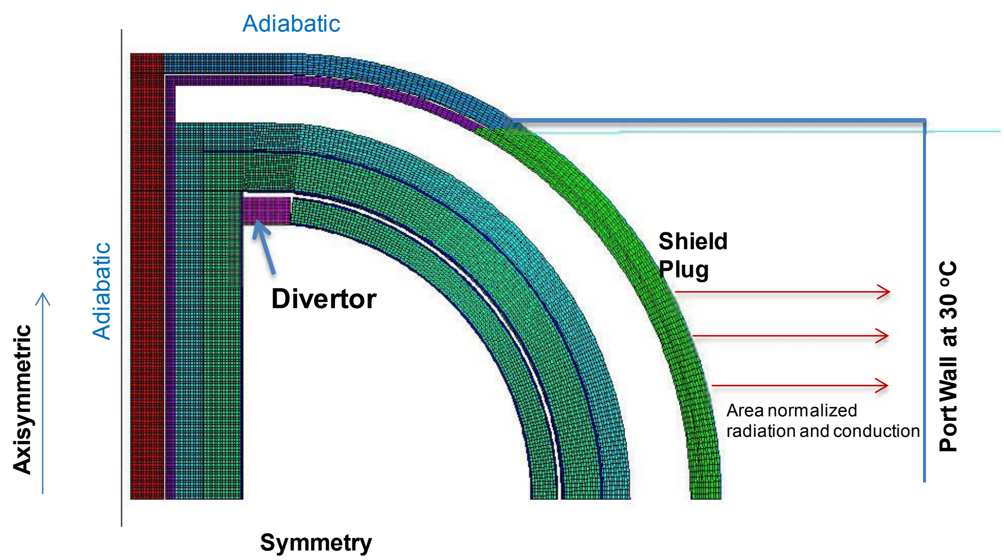


Fig. 6. Axisymmetric finite element model of the power core.

Decay heats were calculated for discrete heating zones in the various components. These heats were applied as time dependent volumetric heating over the course of one year. Examples of the decay heat in various components are illustrated in Fig. 7 for the outboard region and Fig. 8 for the inboard region. Note that for the inboard region, decay heats are plotted for a WC LT shield and a B-FS LT shield. The WC LT shield data correlates to the radial build of Fig. 4, and the B-FS LT shield data is for a design modification illustrated in Fig. 5. The first wall has the highest decay heating. The average first wall decay heat is over 1 $MW/m^3$ immediately after shutdown of the plasma, but drops over an order of magnitude within one hour – a unique feature of SiC. As mentioned previously, plasma energy remains on for 3 seconds following the onset of loss of coolant/flow. During this time, predicted fusion heating values are applied to the various components. Initial temperatures at the onset of LOCA/LOFA are applied to the various structural components and coolants based on the expected operational temperatures. These initial values are summarized in Table I.

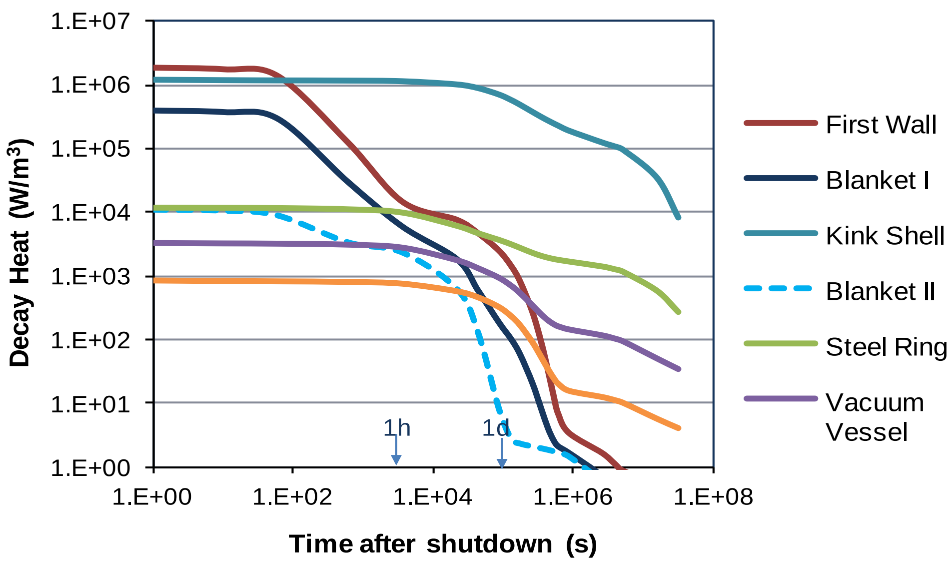


Fig. 7. Decay heats in outboard components.

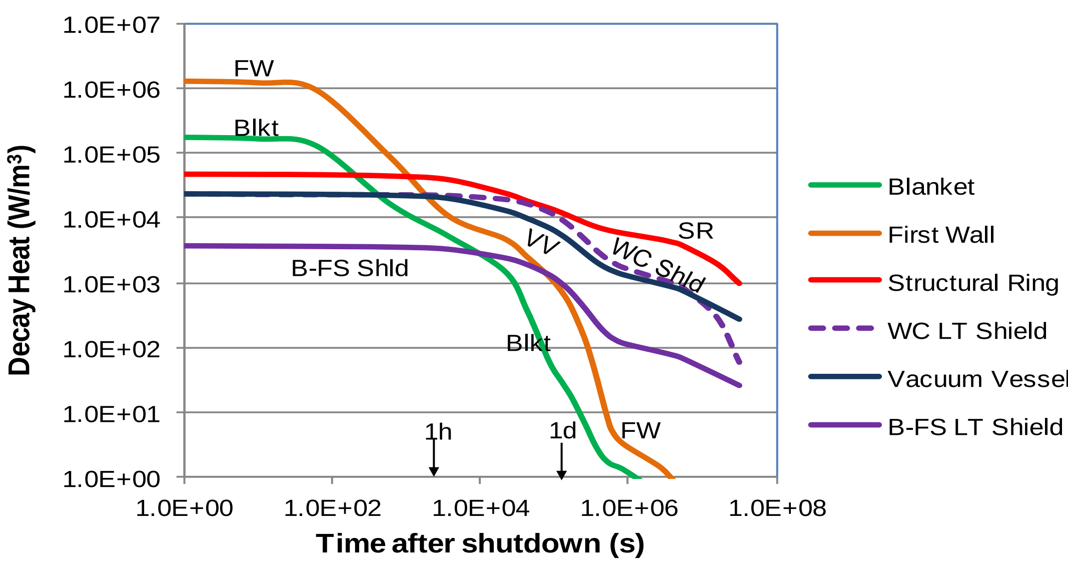


Fig. 8. Decay heats in inboard components.

Table I. Temperatures at Onset of LOCA and LOFA

| Component | Initial Temperature °C |
|---|---|
| First Wall | 900 |
| Blanket | 850 |
| Structural Ring | 675 |
| Vacuum Vessel | 300 |
| Shield Plug | 300 |
| LT Shield | 25 |
| Divertor Plates | 800 |

## III. RESULTS

The first sets of results include the assumption that radiation across the plasma channel could be neglected. The temperatures in the various components are plotted for the outboard side in Fig. 9 and for the inboard region in Fig. 10. It should be noted that the temperatures plotted are for points on the symmetry midplane of the model. This will be the hottest location on the inboard side, but temperatures on the outboard side are higher away from the maintenance port. On the outboard side, temperatures in the structural ring reach 800°C, exceeding the 750°C reusability limit for the ferritic steel. Temperatures on the inboard side are much hotter with maximum temperatures approaching 1750°C occurring after several weeks. This indicates that conduction rates through the core structure are not high enough to remove heat from the inboard regions to the maintenance ports. Examining the first wall temperatures, there are large differences between the inboard and outboard first wall temperatures most notably at times greater than one day. This would indicate that significant radioactive heat transfer would occur across the plasma channel, and our initial assumption on this is invalid. The model was then modified to include an approximate model for radiation across the plasma channel. The analysis was repeated with cross channel radiation included and the results are shown in Figs. 11 and 12. The maximum inboard temperature is reduced to 1007°C while the maximum temperature in the outboard structural ring is virtually unchanged. This indicates that cross channel radiation is much more effective than conduction around the periphery of the power core. These results assume an emissivity of 0.2 for all metallic surfaces and 0.9 for blanket silicon carbide surfaces. The metallic surfaces could be treated to improve their radioactive properties. Because radiation has been shown to be such a strong driver, a study was made to see how emissivity affects the computed structural temperatures. The surface emissivity for all surfaces was varied from 0.1 to 0.9 and the maximum temperatures for the hottest inboard LT shield and the outboard structural ring are plotted in Fig. 13. Maximum temperatures are seen to decrease with increasing emissivity and maximum outboard temperatures fall below 750°C, but inboard temperatures still remain over 900°C. It appears design modifications would be required for reusability temperature limits to be met for such an accident.

Because the maximum temperatures occur in the inboard LT shield and the WC in the shield has a relatively higher and more persistent decay heat rate as shown in Fig. 8, a modification was made to the inboard LT shield where the WC was replaced by B-FS. Using B-FS filler reduces the shield decay heating by more than a factor of 6. The reduced shielding capability of B-FS as compared to WC does require an increase in the shield thickness from 33 cm to 37 cm (refer to Figs. 4 and 5). This modification was incorporated into the FE model and the analysis repeated. For the case with a metallic emissivity of 0.2 and radiation across the plasma channel (comparable to the results shown in Figs. 11 and 12), the maximum temperature in the inboard LT shield was reduced from 1007°C to 965°C as illustrated in Fig. 14. Maximum temperatures on the outboard side were largely unaffected. A further analysis was performed with all emissivities set to 0.9 which would represent a best case heat removal situation for a LOCA/LOFA in the tokamak. The outboard, inboard, and divertor plate (44.9% W and 11.1% ODS-FS) temperature results for this case are shown in Figs. 15, 16 and 17. The outboard structural ring temperature stays below 735°C in this case, but the inboard LT shield and divertor temperatures reaches 891°C and 895°C, repectively, exceeding the reuse temperature of FS by 140-150°C. This means the inboard FS-based components and divertor should be replaced after such a severe accident. However, more advanced NFAs allow remarkably higher reuse temperatures, potentially to 1050°C.[7,8] Therefore, replacing the FS with an NFA in the structural ring, VV, LT shield, and divertors would allow continued use of these systems as long as transient temperatures during an accident remain below 1050°C.

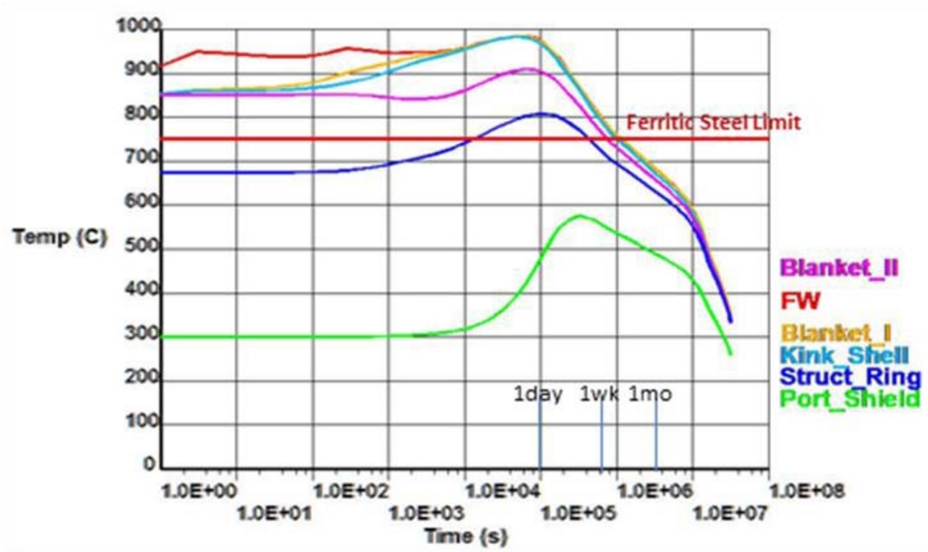


Fig. 9. Transient temperatures at midplane of outboard components after LOCA without cross-channel radiation modeled.

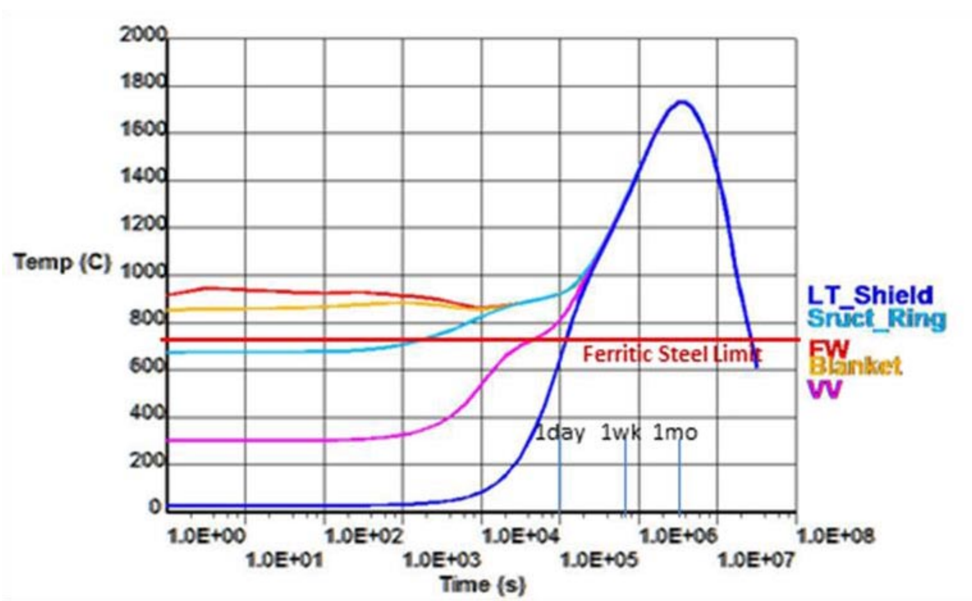


Fig. 10. Transient temperatures at midplane of inboard components after LOCA without cross-channel radiation modeled.

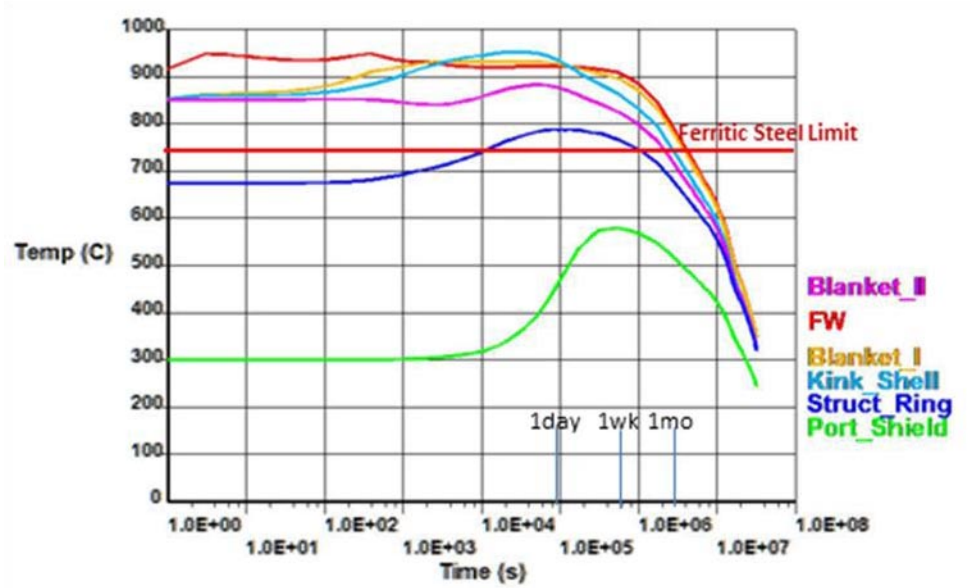


Fig. 11. Temperatures at midplane of outboard components after LOCA with cross channel radiation included.

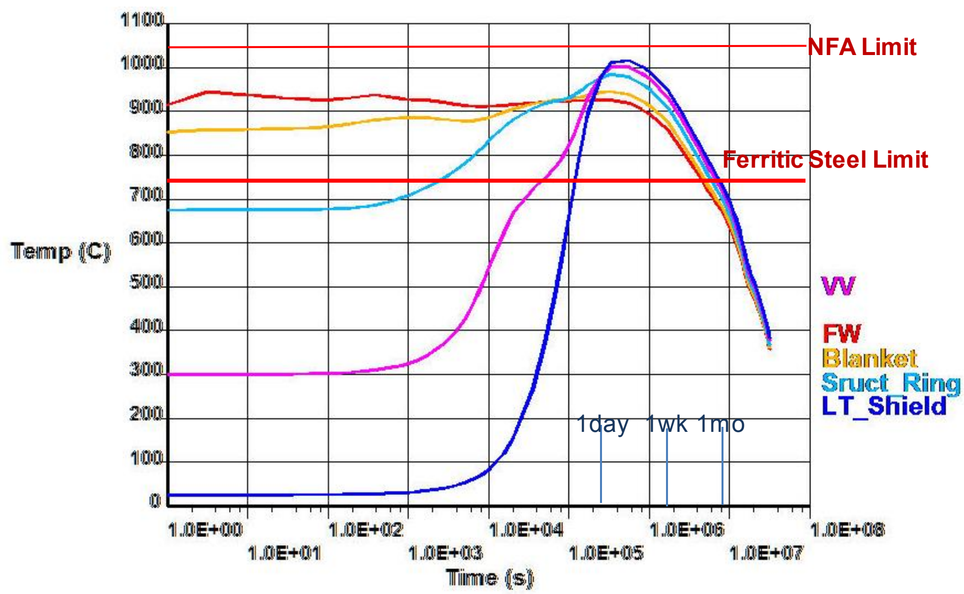


Fig. 12. Temperatures at midplane of inboard components after LOCA with cross channel radiation included.

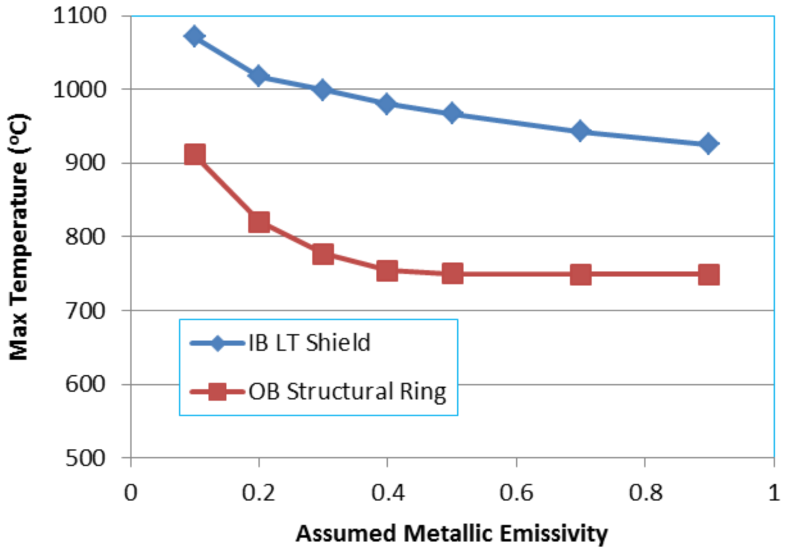


Fig. 13. Effect of assumed emissivity of the metallic surfaces on inboard and outboard temperatures of FS-based components.

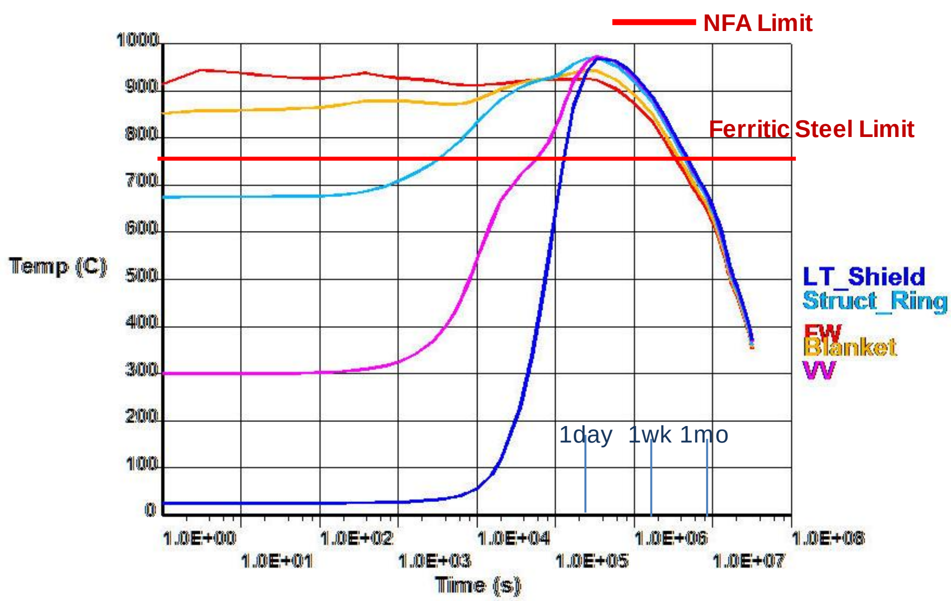


Fig. 14. Inboard temperatures with B-FS inboard LT shield and assumed 0.2 emissivity for the metallic surfaces.

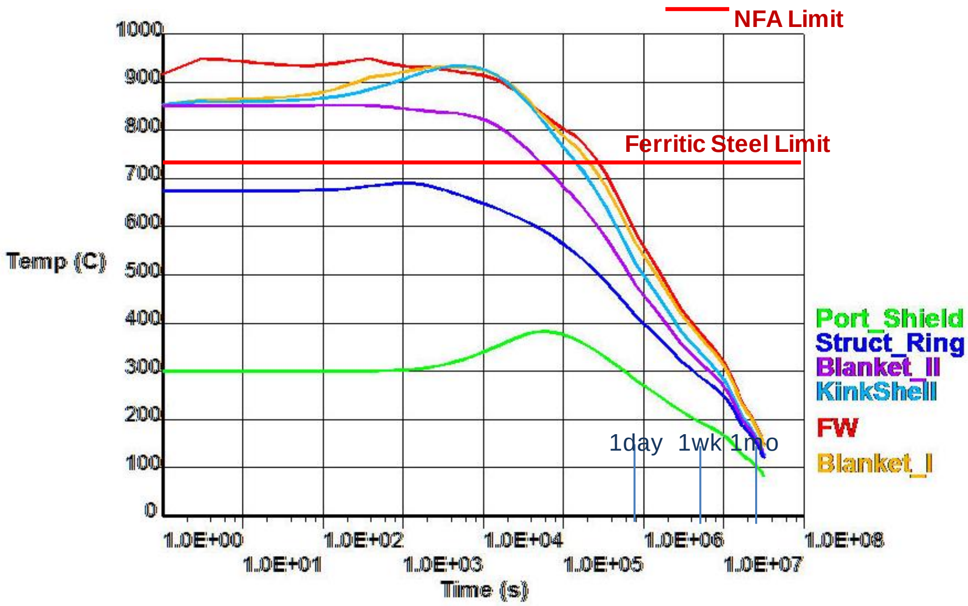


Fig. 15. Outboard temperatures with B-FS filler in inboard LT shield and assumed emissivities of 0.9 for all surfaces.

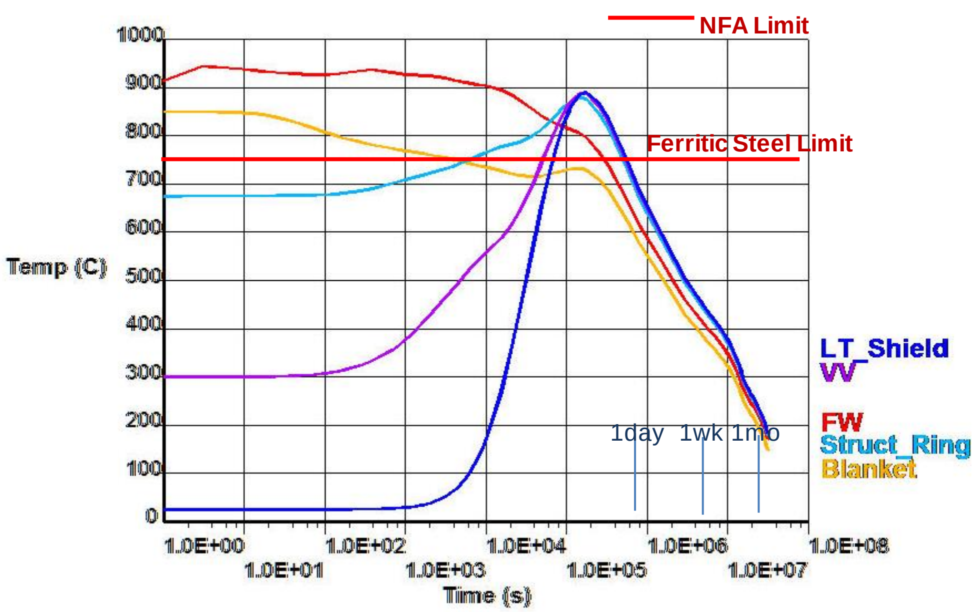


Fig. 16. Inboard temperatures with B-FS filler in inboard LT shield and assumed emissivities of 0.9 for all surfaces.

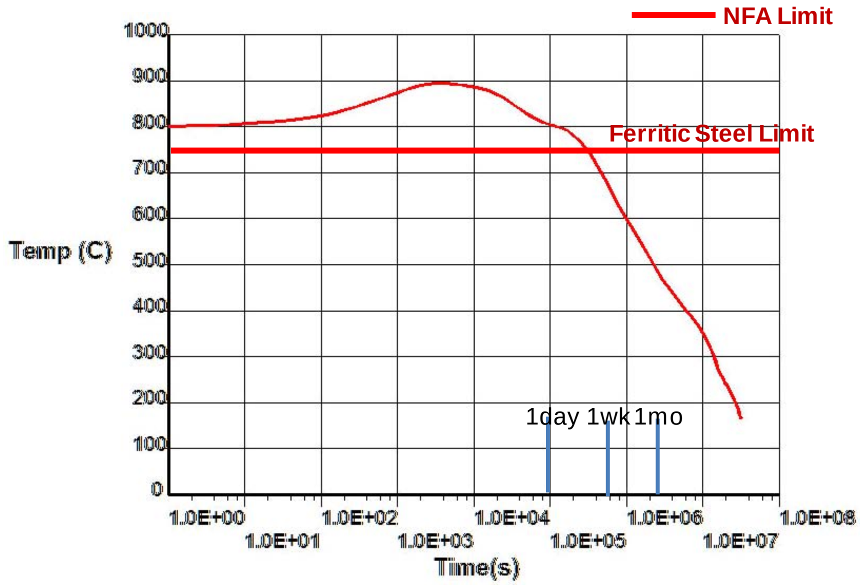


Fig. 17. Bulk divertor plate temperatures (emissivities of 0.9 for all surfaces).

## IV. CONCLUSIONS AND DISCUSSION

The thermal response to a loss of coolant/flow accident has been modeled for the ARIES-ACT-1 fusion power plant using finite element analysis techniques. These transient analyses simulated a LOCA for the helium and water coolant loops and loss of LiPb flow for up to one year following the event. The analyses assumed no heat loss through the superinsulation between the LT shield and magnet. Thus, the only avenue for removal of heat from the power core is through the outboard maintenance ports. The design results in significant heat buildup in the inboard sections of the power core with temperatures exceeding the 750$^o$C projected ferritic steel reuse temperature. A design modification that changed the inboard LT shield filler from WC to B-FS helped reduce the inboard temperatures, but they still exceeded the 750$^o$C reuse temperature by 141$^o$C. Improvements in modeling and assumptions, such as more inclusion of heat losses through the piping and outboard penetrations would likely result in a decrease in calculated inboard temperatures, but it is doubtful that inboard and divertor ferritic steel temperatures would drop below 750$^o$C. Thus, if such a severe accident is considered to be part of the design basis, it is essential to either improve the inboard heat removal mechanism during LOCA/LOFA, use more advanced structural materials (such as Nano-structured Ferritic Alloys) in the inboard LT-shield, VV, SR, and divertor, or replace all four components after such a severe accident.

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