ARIES-CS Loss of Coolant and Loss of Flow Accident Analyses

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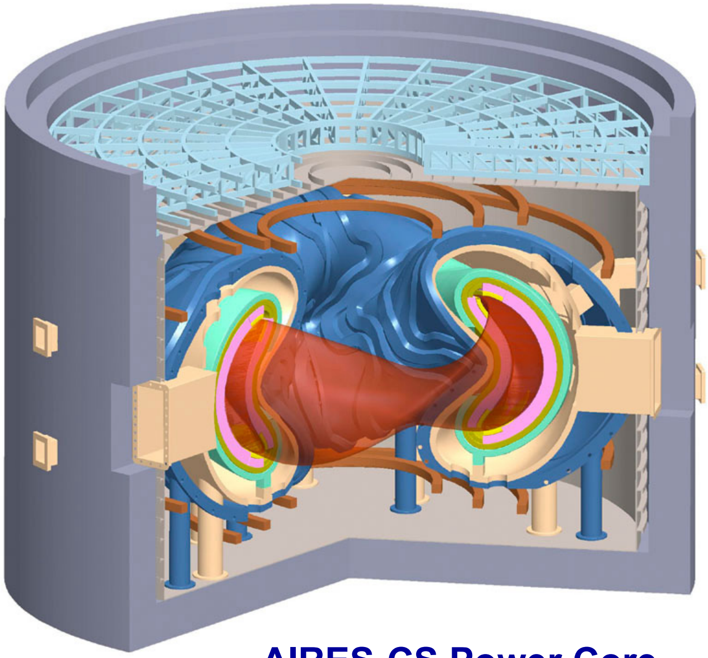
Introduction and Objectives

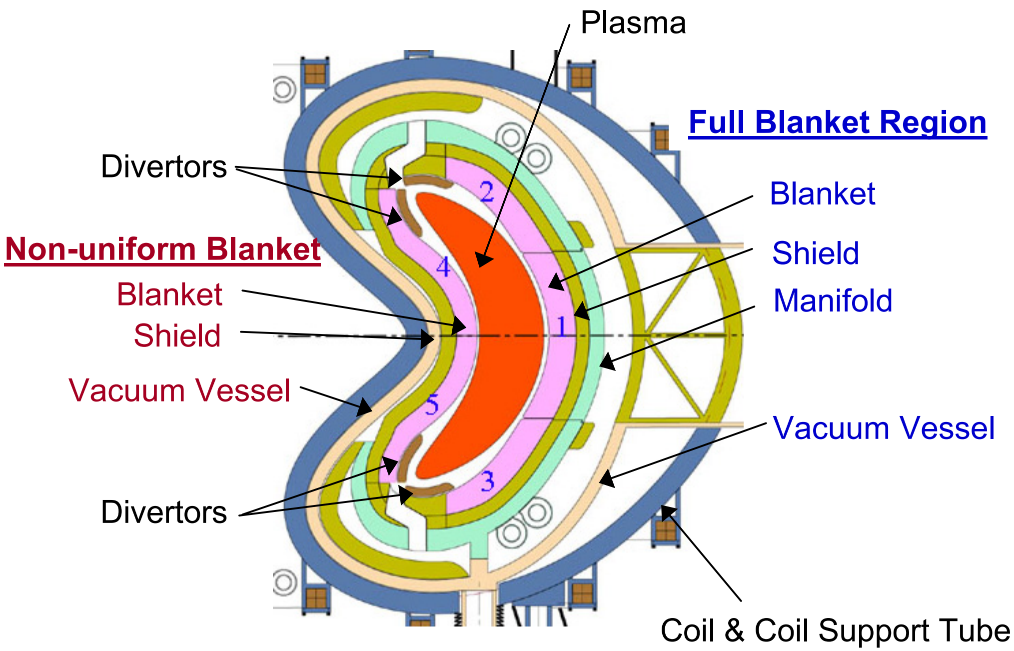
- ARIES-CS is a 1000 MW fusion power plant conceptual design launched to study the physics and engineering of compact stellarator (CS) power plants.
- Loss of Coolant Accident (LOCA) and Loss of Flow Accident (LOFA) thermal simulations for ARIES-CS are described in this presentation.
- The presence of three coolants: helium (He), lithium lead (LiPb), and water (H_2O), makes multiple LOCA and LOFA combinations possible.
- Two LOCA/LOFA combinations have been investigated: LOCA for helium and LOFA for LiPb and water, and a beyond design basis accident of LOCA for helium and LiPb and LOFA for water.
- Transient finite element analysis is used to determine if temperatures produced by decay heat exceed the reuse temperature of the power core components.

ARIES-CS Configuration

ARIES-CS (compact stellarator) power plant design



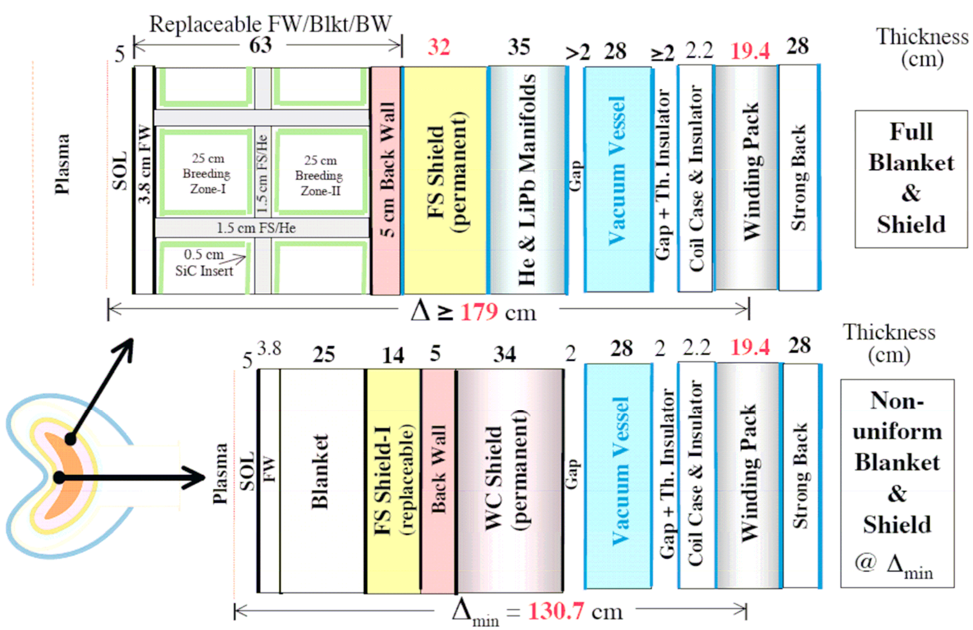
AIRES-CS Power Core

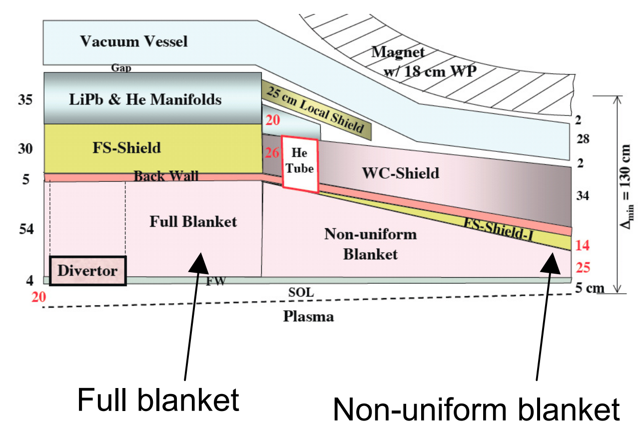


0 Degree Cross-section

Blanket Concept Radial Builds

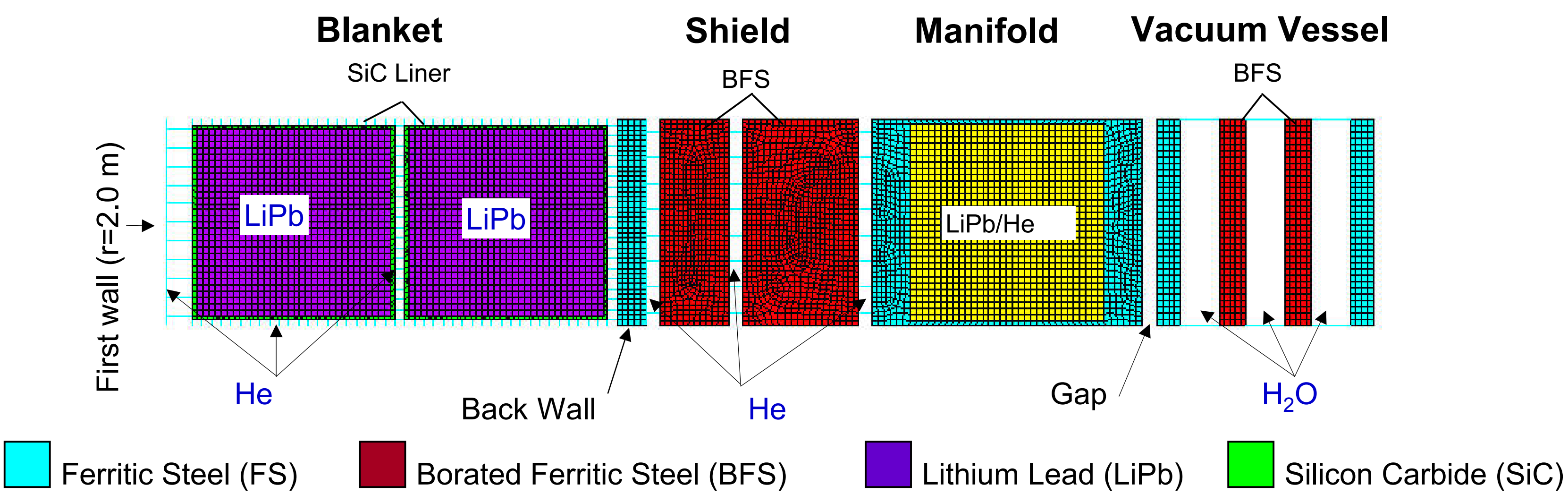
- Full Blanket design covers 61% of total surface area.
- Non-uniform blanket used over 24% of surface where clearance between plasma channel and coil prevents use of full blanket.
- Remaining 15% of area covered by divertor plates.





Full Blanket Finite Element Model and Design Details

- Modular blanket uses three coolants: helium (He), lithium lead (LiPb), and water (H_2O)
- Three materials used in the power core: ferritic steel (FS) for most structural parts, borated ferritic steel (BFS) as a filler in the shield and vacuum vessel (VV), and a silicon carbide (SiC) thermal and electrical liner in the blanket.
- The design repeats in the vertical direction in the figure below allowing for use of symmetry on the top and bottom planes. Modeled as axisymmetric about the plasma channel.



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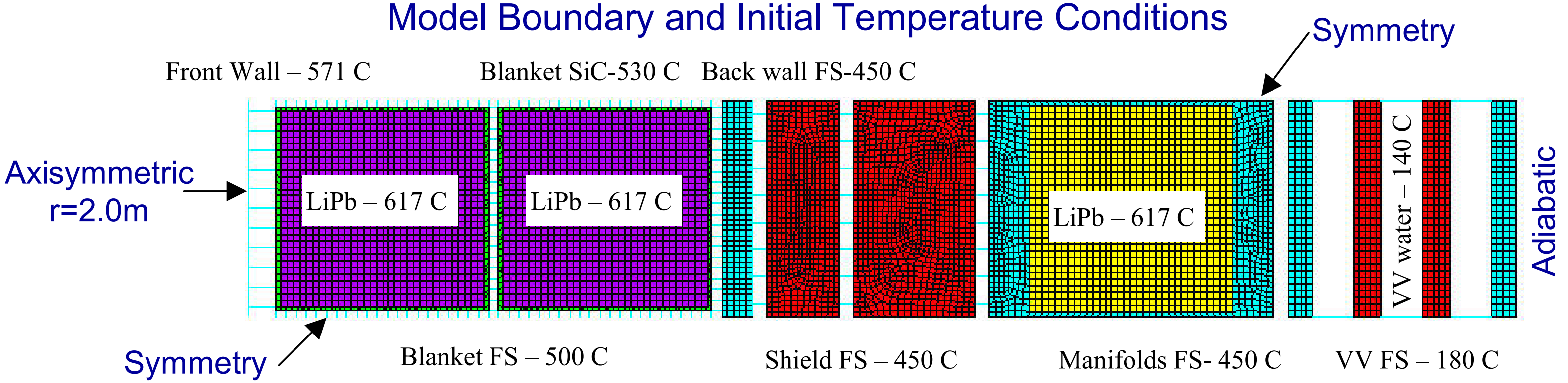
LOCA/LOFA Combination for Various Coolants

- Only loss of flow condition considered for water in vacuum vessel. Lower temperature water coolant loop is considered to have extremely high reliability and natural convection in loop provides only avenue for heat removal from power core.
- Only loss of coolant considered for helium in blanket and shield. This is considered worst case for He, but LOCA/LOFA probably not significantly different for He.
- For the LiPb, loss of flow is the design base line accident. The time required to drain the LiPb coolant (hours) makes a sudden loss of coolant unlikely.
- Most emphasis is thus placed on LOCA in the helium and LOFA in the water and LiPb which represents the most likely, most serious accident scenario.
- A beyond design base line accident which includes loss of LiPb coolant is also presented briefly although this condition is highly unlikely.
- LOCA/LOFA conditions are assumed to exist in the entire power core even though the highly modular nature of the ARIES-CS makes this unlikely.

Analysis Input and Assumptions

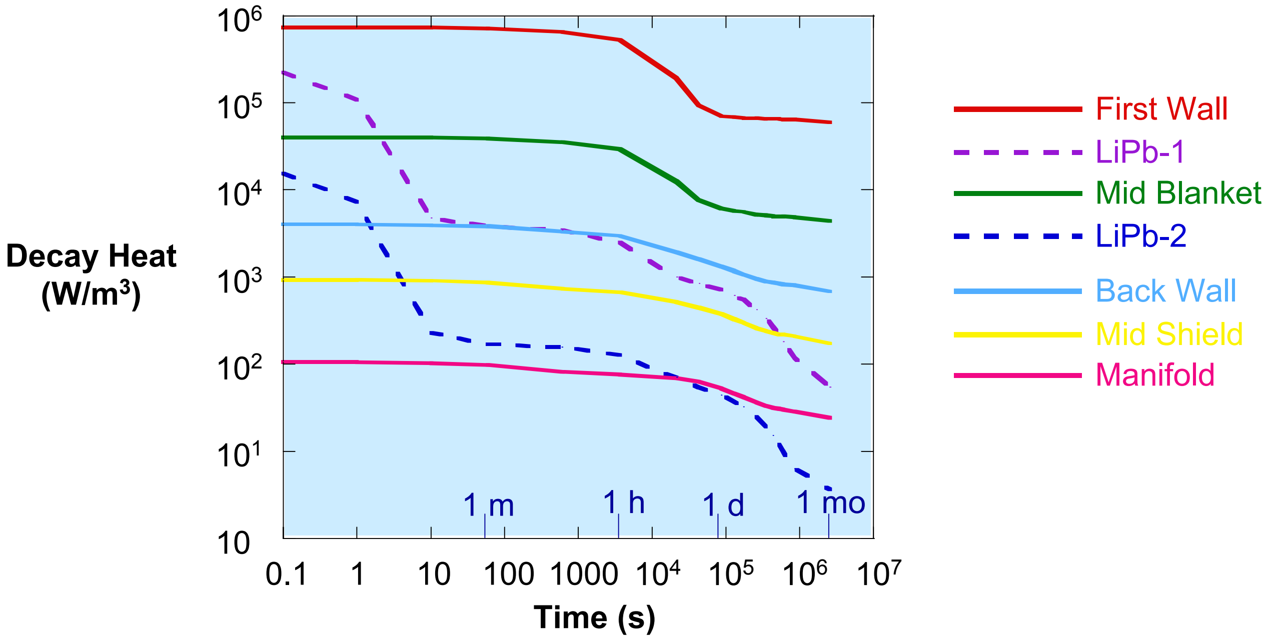
- The plasma is assumed to remain on for three seconds after onset of LOCA/LOFA.
- Nuclear loads and decay heats are based on 2.6 MW/m^2 average neutron wall loading.
- The back wall of the VV is assumed to be adiabatic because of the significant thermal insulation between the VV and the cryogenic coils and coil support tube.
- Radiation occurs between parallel surface in the empty coolant channels and the gap with an emissivity of 0.3.
- Water remains in the vacuum vessel at 140°C and circulates naturally with a convection coefficient of 500 W/m^2-C.
- The LiPb does not circulate.

Model Boundary and Initial Temperature Conditions



Full Blanket Decay Heats

- Decay heats calculated for 20 discrete heating zones and 18 time points.
- First wall maximum decay heat of 730 kW/m^3 drops an order of magnitude in 24 hours.
- Decay heat levels fall quickly through depth of blanket – maximum back wall heat is 4 kW/m^3.



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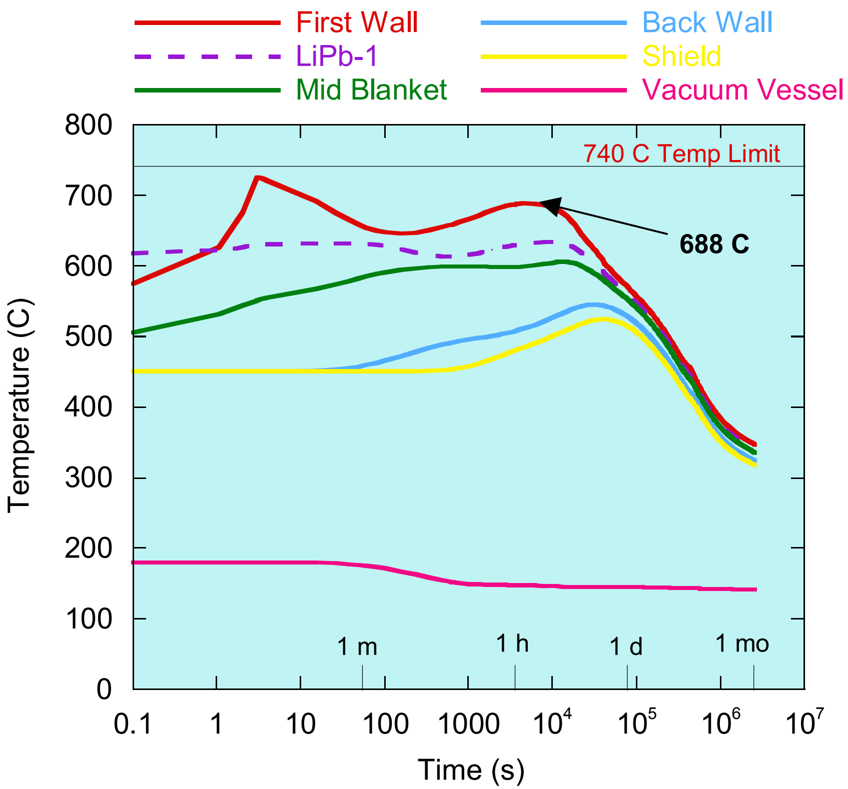
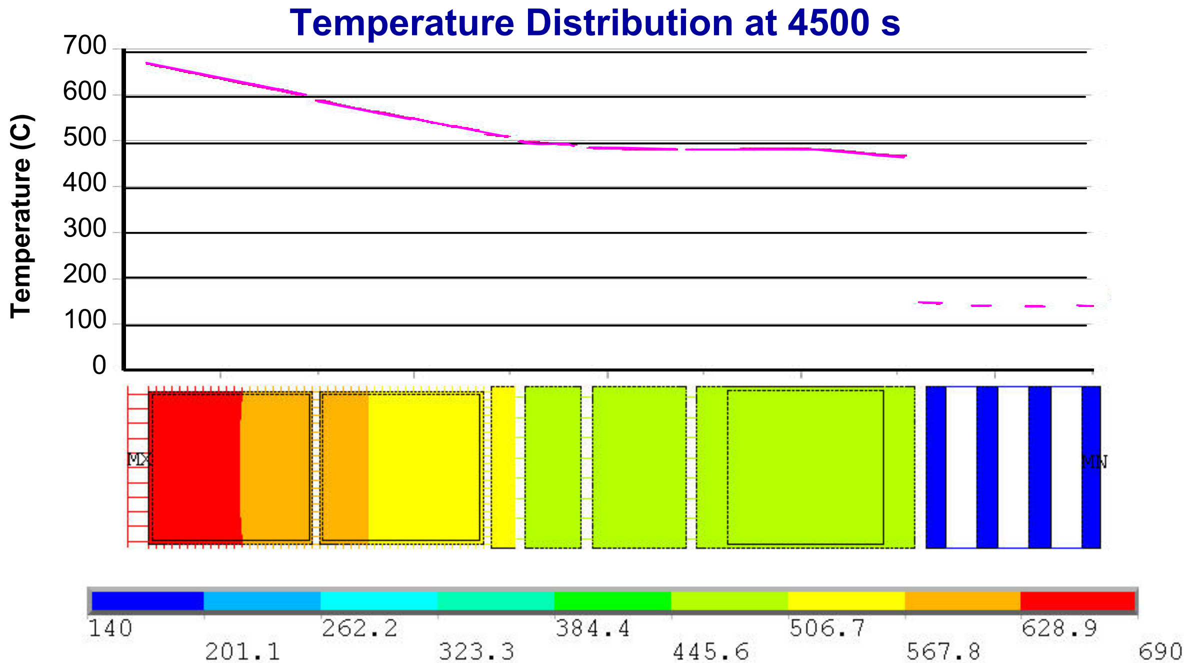
Transient Thermal Response of Full Blanket

- Maximum first wall temperature, 725°C, occurs immediately after shutdown of plasma at 3 s.
- Second temperature peak (688°C) occurs at 4500 s (75 min).
- All component temperatures remain below the 740°C FS reuse temperature.
- Temperatures decline steadily after 12 hrs.



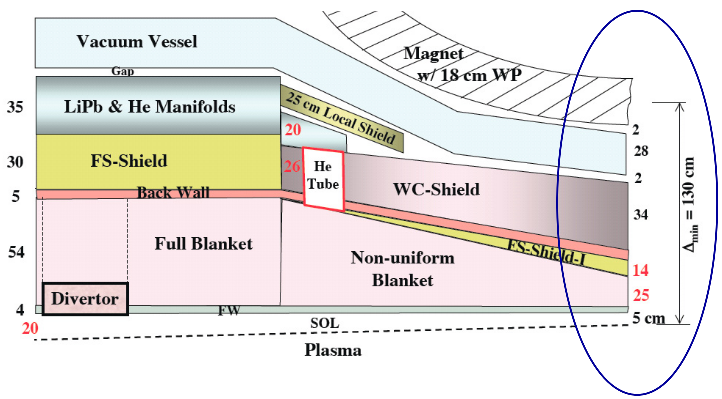
Radial Temperature Distribution

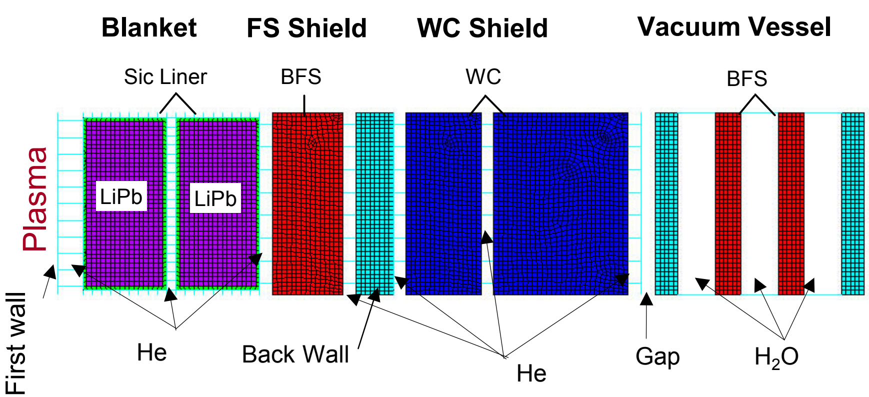
- Large thermal gradient across gap indicates gap limits heat removal making temperatures sensitive to emissivity.
- Small temperature difference between water and VV structure means temperatures will be insensitive to convection coefficient.



Non-Uniform Blanket Finite Element Model and Design Details

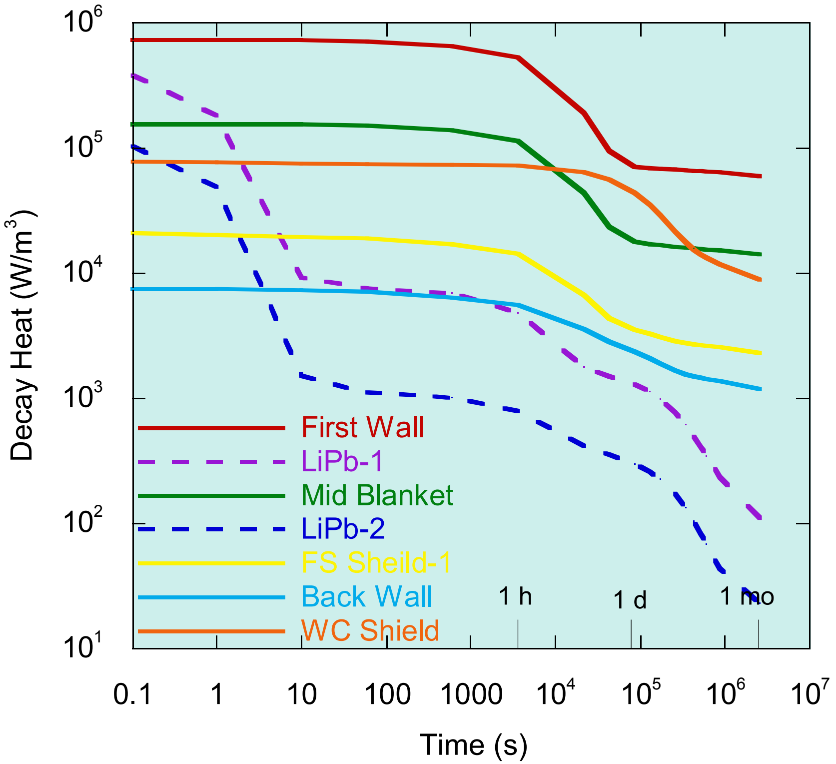
- Design similar to full blanket with reduced thickness blanket and FS-shield. Major difference is addition of tungsten carbide (WC) shield.
- Vacuum vessel design is identical to full blanket.
- Minimum thickness blanket modeled – transition region not considered.





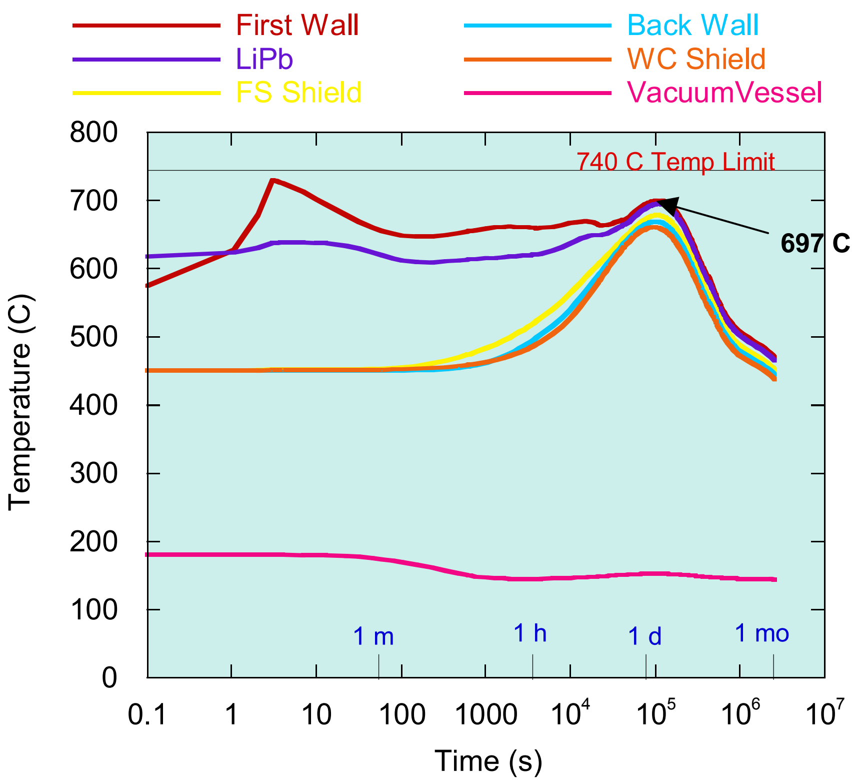
Non-Uniform Blanket Decay Heat and Analysis Assumptions

- Decay heats similar to full blanket but slightly higher due to reduced breeding area.
- Significant difference is the relatively high decay heat in WC shield.
- Analysis assumes blanket section is completely surrounded by non-uniform blankets, i.e. no lateral heat conduction or radiation across plasma channel.
- Initial and boundary conditions same as full blanket model.



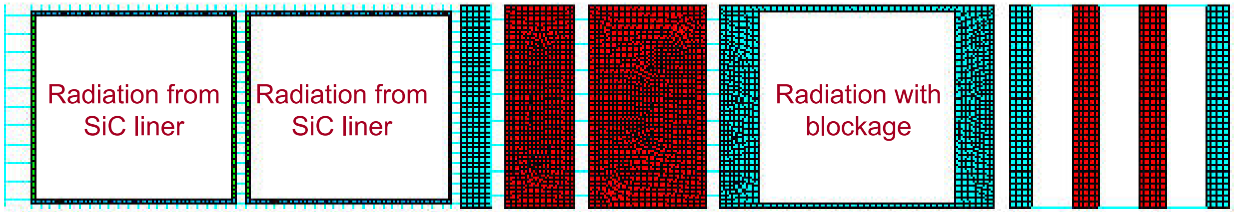
Transient Thermal Response of Non-Uniform Blanket

- Maximum first wall temperature, 730°C, occurs immediately after shutdown of the plasma (3 s).
- Second first wall temperature peak (697 °C) occurs over 24 hrs. after incident.
- All component temperatures higher due to increased heating in WC shield.
- Again all component temperatures remain below the 740°C FS reuse temperature.
- All temperatures decline long term.



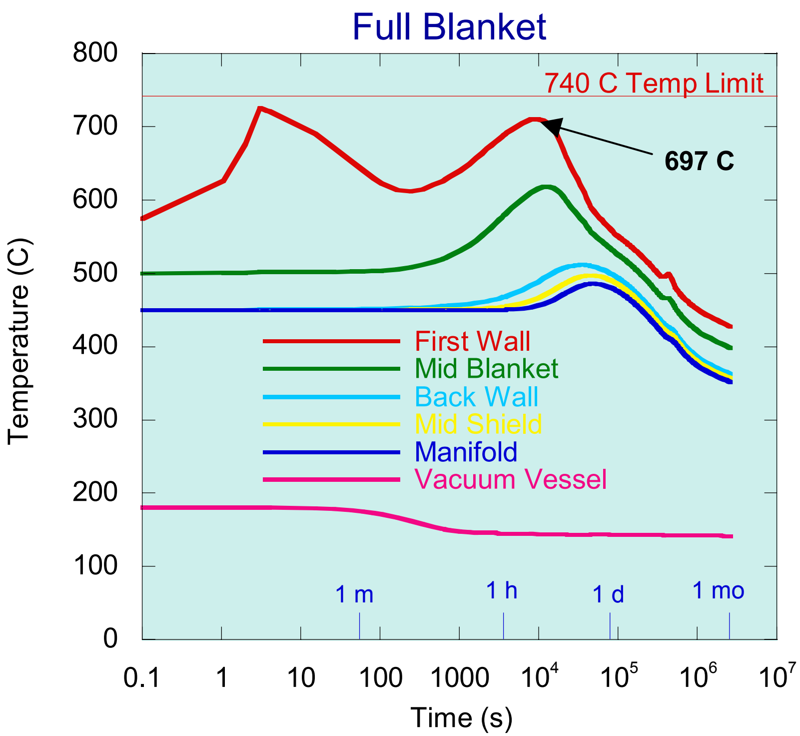
Beyond Design Basis Accident LOCA in LiPb and He

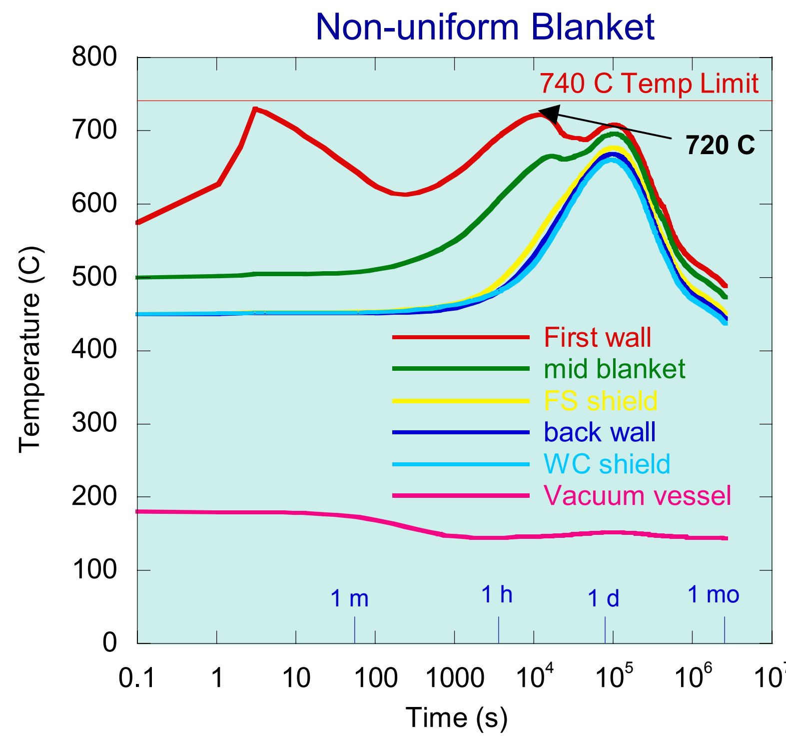
- LOCA for LiPb considered highly unlikely. System drainage time is on the order of hours so immediate loss of both He and LiPb would require catastrophic event.
- Analysis does provide insight into longer term thermal behavior of power core with LiPb and He removed.
- In model, LiPb is removed from blanket and manifold and replaced with radiation transfer elements.
- A reduced SiC emissivity value of 0.5 is assumed in the blanket LiPb cavity to account for contact issues between the FS wall and SiC liner.
- Effective radiation area in manifold reduced to account for blockage from internal plumbing.



Results for LOCA in LiPb and He

- Temperatures slightly higher for LiPb LOCA but remain below reuse temperature.
- Results sensitive to emissivity in LiPb passage. An emissivity of 0.3 produces a maximum temperature of 753 ^{o}C for the full blanket.





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Conclusions

- Transient finite element analysis of LOCA/LOFA events for the ARIES-CS power plant design indicate temperatures remain below the 740 °C ferritic steel reuse temperature.
- Structures remained below reuse temperature for the extreme design basis case of LOCA for He and LOFA for LiPb and water and also for a beyond design basis case of LOCA in both LiPb and He.
- Maximum temperatures (725-730 °C) occur immediately after shutdown of plasma 3 seconds after LOCA/LOFA incident. The assumptions involved in plasma shutdown time and thermal loadings are critical to this response.
- The analyses are predicated on heat being removed from the power core by natural convection to water in the vacuum vessel. Results not sensitive to assumed convection coefficient.
- Future work will examine LOCA/LOFA performance of divertor system.