NENE 2022
September 12-15
PORTOROŽ
SLOVENIA
31st International Conference Nuclear Energy for New Europe

Natural Convection Cooling in DEMO Vacuum Vessel during EX-VV LOCA

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ABSTRACT

This study evaluates the temperature evolution of DEMO in-vessel components (IVCs) when cooling is impaired during an ex-vessel Loss of Coolant Accident (LOCA), including the 120s-long plasma ramp-down. It is conservatively assumed that a large (guillotine) break in the main feeding line of the breeding blanket (BB) PHTS causes an instant loss of cooling in all BB segments.

The boundary conditions and decay heat data have been set according to the latest update of DEMO design and conditions. The design upgrade includes the BB supports that are attached to the actively cooled Vacuum Vessel (VV) wall. Concerning the heat transfer model, the radiation heat transfer is modelled explicitly using the S2S radiation model. A transient CFD analysis with the ANSYS Fluent code has been carried out to assess the temperature distribution of the in-vessel components and the redistribution of the thermal loads due to in-vessel natural convection of injected helium, heat conduction through supports and radiation heat transfer.

The ex-vessel LOCA analysis described here shows that natural convection with filled gas seems to provide a sufficient cooling mechanism for decay heat removal from BB segments even in case of an instant loss of cooling in all BB segments. The peak temperature in the considered water-cooled lead lithium (WCLL) BB segments occurs within the first two hours from the onset of the event and reaches approximately 550°C. The distribution of thermal loads in the BB segments shows that the natural convection cooling of the filled gas is an important contributor to decay heat removal.

1 INTRODUCTION

The DEMOnstration fusion power reactor (DEMO) is identified as a key step following ITER towards the exploitation of fusion power [1]. As such, DEMO must demonstrate the adequacy of implemented technologies for safe electricity production, developed on consideration that regular, rapid and reliable maintenance strategy of the plant can be carried out remotely [2]. With main requirements of DEMO to achieve operation with a closed fuel-cycle (i.e. tritium self-sufficiency) and reliable long-pulse plasma operation [3], the majority of the fusion energy transported by neutrons has to be absorbed in the breeding blanket (BB) [4]. Very high neutron fluence (more than one order of magnitude higher than in ITER [3]) will result in a high irradiation of in-vessel components (IVC) and high decay heat after the plasma shutdown. In [5] it is shown that during in-vessel maintenance, i.e. after a cool-down period of 4-8 weeks, the decay heat of the BB is reduced sufficiently to avoid excessive temperatures that would impede the remote handling operation. In this article the assessment of the accidental scenario is presented when the BB segments would lose their active cooling during plasma operation when the decay heat is significantly higher due to the presence of short-lived isotopes. The assumption that all BB segments would be affected by the LOCA is conservative because it may be considered in the future to implement separate and fully independent BB PHTSs, each cooling only a subset of the BB segments. The sequence of the accident scenario considered in this study is assumed as follows:

- 0s: Ex-vessel guillotine break of BB PHTS main line;
- 0s: Loss of active cooling in BB PHTS (all BB segments);
- 0s: Initiation of plasma-ramp down (linear fusion power decay is assumed);
- 120s: Completion of plasma-ramp down;
- 120s: Reduction of VV temperature by 5°C/h until it reaches 30°C;
- 150s: Initiation of VV venting with helium (not modelled);
- 210s: 95 kPa inside VV

In an ex-vessel LOCA a guillotine rupture of a large coolant pipe of the PHTS outside the Vacuum Vessel (VV) is postulated. Consequently, the active cooling of the Breeding Blanket (BB) segments in all tokamak sectors is lost almost instantly. Due to the decrease of coolant pressure in the BB PHTS, the isolation valves at the external walls of the tokamak building isolate all secondary cooling loops, the main PHTS pumps are switched off and the shutdown cooling system is used to cool the reactor. After the detection of the LOCA a soft plasma shutdown is initiated automatically. Other in-vessel component cooling circuits maintain their nominal flow until the plasma ramp down is completed. After that, it is assumed that the VV temperature is reduced by 5°C/h until it reaches 30°C. The injection of helium starts 30 seconds after the completion of a soft plasma shutdown [8]. Once the VV is filled at a pressure of 95 kPa (i.e. within a minute) the valves are closed and the helium will circulate naturally inside, enabling convective cooling of passive IVCs. Due to the thermal loads during the plasma ramp-down and decay heat, the passive BB segment are heated internally. Eventually, all heat is removed from the reactor by the safety-classified VV active cooling system.

The main objective of this study is to assess the described transient investigating the time dependent temperature distribution in passive breeding blanket segments and to assess the feasibility of the proposed cooling solution by filled gas for residual heat removal. A conjugate heat transfer model is applied to solve the coupled heat transfer in the solid and fluid (circulating gas). Heat transfer model considers also the radiation heat exchange that is modelled with the Surface-to-Surface (S2S) radiation heat transfer model [7]. The ANSYS Fluent code 2021R2 [7] was used to run transient CFD simulations over a time of several hours. CFD input model used in this study includes one sector of the DEMO tokamak. For the purpose of this CFD analysis, the geometry of in-vessel components is simplified. The homogenized material properties are applied to BB segments, with consideration of the Water-Cooled Lithium-Lead (WCLL) concept. This work can be considered as the continuation of previous studies [5],[9] but with refined initial and boundary conditions. The most important updates of the model represent the consideration of refined ("realistic") decay heat loads [10] and refined initial temperatures of IVCs [8].

The structure of the paper is as follows: in Section 2, the CFD simulation setup is briefly presented, the results of analysis are discussed in Section 3 and the main conclusions are drawn in Section 4.

2 SIMULATION MODEL

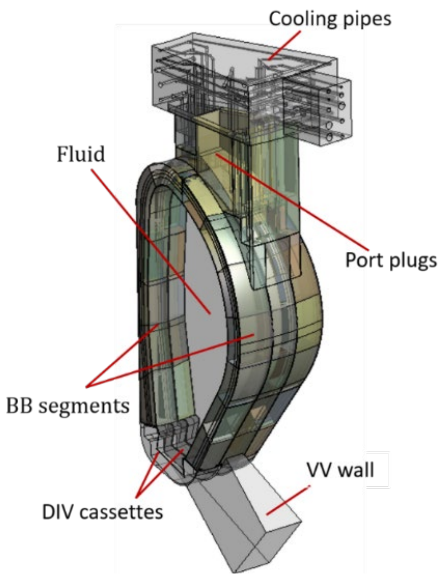
The simulation is started at the beginning of the so-called plasma ramp-down phase, which is assumed to last 120s. As such, the initial (3s long) heating of BB segments when the cooling is already impaired but the plasma shutdown is not yet initiated has not been simulated. However, it has been shown [11] that less than 5°C lower predicted peak BB temperatures are obtained when the so-called response time is not considered in the sequence of the accident.

The inflow of helium into the vacuum vessel is not modelled in this study. Instead it is conservatively assumed that the vacuum in the VV is kept until the end of the foreseen “helium injection phase”. During this time (initial 210 seconds of transient), the cooling of all BB segments is achieved only by the radiation heat transfer.

2.1 Geometry and applied boundary conditions

A recent 2017 geometry of DEMO tokamak with 16 equatorial sectors is adopted for this study. The simulation model, shown in Figure 1 (left), includes one equatorial sector of DEMO tokamak with 5 BB segments. The upper port is closed at the level of the upper port ring channel. The (actively cooled) divertor cassettes are removed from the model for the purpose of this study. As such, the (fluid) wall boundaries with prescribed temperature are placed to represent the external surfaces of the divertor cassettes. Other in-vessel structures, such as supports, chimneys and upper port plugs are considered to be passive, exchanging heat with surrounding fluid and adjacent components. Both side boundaries of fluid domain are modelled with the symmetry boundary condition, whereas the inner walls of VV are modelled as no-slip walls with prescribed temperature. The external surfaces of cooling pipes within the upper port plugs are modelled as the adiabatic no-slip walls.

To better consider the decay heat data, each BB segment is sliced radially and in poloidal direction, which is shown in Figure 1 (right). The applied segmentation is to-the-best-effort possible similar to the one used in the MCNP calculations [10]. Based on the material composition of each of the blanket components [10], homogenized material properties are applied [11].



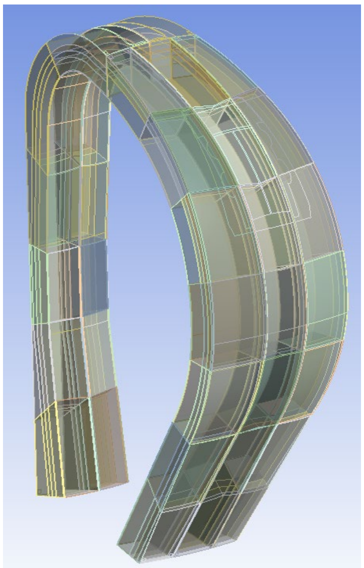


Figure 1: Computational domain with main in-vessel components (left). Segmentation of BB segments (right).

2.2 Numerical Mesh

Numerical meshes are created with the "ANSYS Meshing" tool [6], by using the automatic meshing method with adaptive cell sizing. To avoid generation of spurious cells at external BB surfaces (i.e. fluid/solid interfaces) two meshes were generated. The first mesh comprises only the BB segments. The second mesh includes the fluid domain, chimneys, supports and plugs (i.e. all other domains). Both generated meshes are conformal within the domains of individual meshes, whereas the mesh at the interfaces between both generated mesh files is non-conformal. Once both meshes have been imported into the Fluent software [7], the initially fully tetrahedral mesh of BB segments is converted to polyhedral mesh. The resulting mesh, which is used in the simulations is somewhat hybrid – BB segments are meshed with polyhedral cells, whereas the other domains/zones are meshed with solely tetrahedral elements.

The mesh generation was controlled by specifying the global mesh element sizes for individual domains. In addition, the fluid mesh is additionally refined in gaps around BB segments using the proximity control option. Proximity defines the level of refinement in small gaps, e.g. in channels between BB segments, BB and VV walls. Four cells are applied in all gaps, as suggested in [7]. The applied mesh parameters are summarized in Table 1.

Table 1: Overview of mesh parameters

Component	Global el. Size	Number of cells	Number of cells after conversion
Plugs	0.3/0.1* m	4,975,263	Not affected
Chimney	0.75 m	1,300,293	Not affected
Supports	3x3x3 divisions	486	Not affected
Fluid	0.5 m	64,454,067	Not affected
BB	0.5 m	45,182,687	9,743,040
Total		115,912,796	80,473,149

* Smaller elements are needed to obtain sufficient mesh quality for the central plug

Snapshot of the "final" mesh, visualized in horizontal cross-section plane is presented in Figure 2. The mesh is refined in all (2 cm wide) gaps to achieve four tetra elements across the gap thickness. Such regions occur in gaps between the adjacent BB segments, individual BB segments and VV walls, between upper port plugs etc.

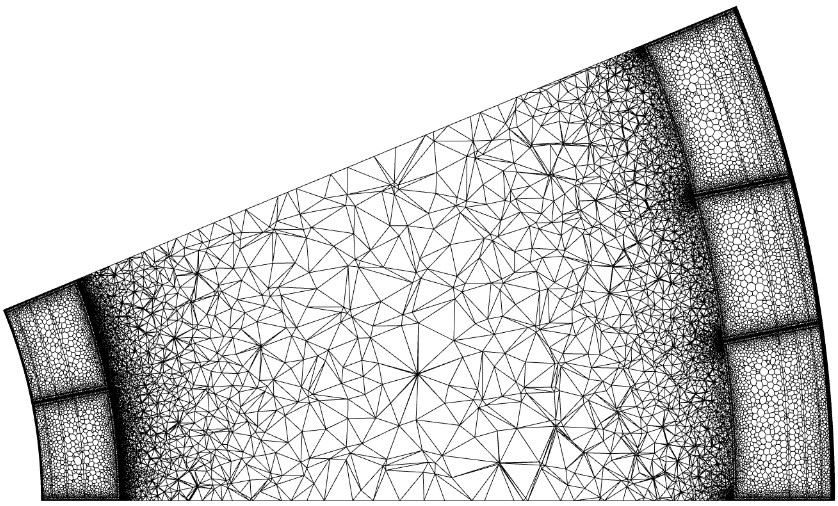


Figure 2: Horizontal slice through the mesh

2.3 CFD code and setup

Transient simulations of the natural convection of filled gas (helium) inside the plasma chamber, governed by the nuclear heat in passive BB segments, have been performed with the

ANSYS Fluent code v2021R2 [7]. The flow is considered laminar [5], [11]. The radiation heat exchange is modelled with the Surface-to-Surface (S2S) radiation heat transfer model [7]. Simulations were run using the PISO algorithm and fixed time stepping method [7]. Additional settings comprise the so-called frozen-flux formulation, pressure scheme called body-force, neighbour and skewness correction. The second order implicit scheme was adopted for time integration of governing equations. More details about the setup are available in [11].

The inflow of helium into the vacuum vessel is not modelled, instead it is conservatively assumed that the vacuum is kept inside the VV for additional 210 seconds after the LOCA occurrence [8]. In the CFD code, the vacuum is modelled by setting the thermal conductivity of fluid to zero (e.g. to 10^{-12} W/(mK)). During this time, only heat transfer equations (including radiation) were being solved, using a fixed time step set to 0.5 s. Once the simulation time reaches 210 seconds (i.e. 120 seconds of plasma ramp-down phase, delay of 30s followed by additional 60 seconds for helium injection), the fluid pressure and temperature have been patched to 95 kPa and 50K, respectively [8]. The simulation shows [8] that the helium average temperature stabilizes at ~270°C after 600s of the initial heat up, while the pressure in the VV remains constant at ~95 kPa.

In order to achieve the solver convergence once the helium is present in the VV, when the natural convection is rather weak, sufficiently small simulation time step is needed. The simulation is initiated with the time step dt set to $1\ \mu$s, which is gradually increased to 0.05 s [11]. To speed up the solution of the transient, the so-called frozen-flow treatment has been used [11]. In practice this means that the full set of equations (Navier-Stokes equation and heat transfer equations) was solved for 25 time steps, then the flow field was frozen for the next 75 time steps when only heat transfer equations were solved. Afterward, the full set of equations was solved again for the following 25 time steps. These steps were automatically repeated until the simulation was stopped. In the case of frozen-flow field, the simulation time step was increased from 0.05 s to 1.0 s, which enables much faster progression of the transient simulation in physical time. Such approach has been tested and verified against the solution obtained without the frozen-flow assumption [9], [5].

The radiation heat exchange is modelled with the Surface-to-Surface (S2S) radiation heat transfer model [7]. Emissivity of steel or tungsten surfaces of VV, divertor and blanket is equal to 0.3 [8]. Radiation iteration in solver is obtained every 10^{th} energy iteration, whereas the maximum number of radiation iterations was increased to 15. The residual convergence criteria for S2S model was set to 10^{-3}. In all computations, up to 14 iterations were needed to obtain the convergence of radiosity. The applied initial temperatures of structural materials [8], used in this study, are given in Table 2.

Table 2: Initial conditions - Components' average temperatures prior to the ex-VV LOCA

Layer/Component	Initial temperature [°C]
FWA, FW	380
Side plates	300
Breeder zone, LiPb manifolds	450
H_2O manifolds, BSS	300
Divertor PFCs/Cassette body	400/200
VV	50

2.4 Thermal loads of BB segments (decay heat data)

The nuclear heat data from [10] is adopted for this study. It should be noted that the 0.5 mm thin front wall armour (FW) is not modelled explicitly in the CFD model. Instead, its nuclear heating is applied by the boundary condition at the FW/fluid interface as volumetric energy source where layer thickness and material properties are specified. More information is available in [11].

3 RESULTS

CFD simulation has been run for 6 hours of physical time, which is sufficient to show that established natural convection together with radiation heat transfer provides sufficient decay heat removal that BB segments eventually start to cool down. Time evolution of peak BB temperatures for the central outboard segment are presented in Figure 3. It is noted that the peak temperature in the BB segments occurs about two hours after the onset of the accident, when the maximum temperature reaches approximately 550°C. The hottest region of the BB segment is the breeder zone, which is in contact with the front wall (FW).

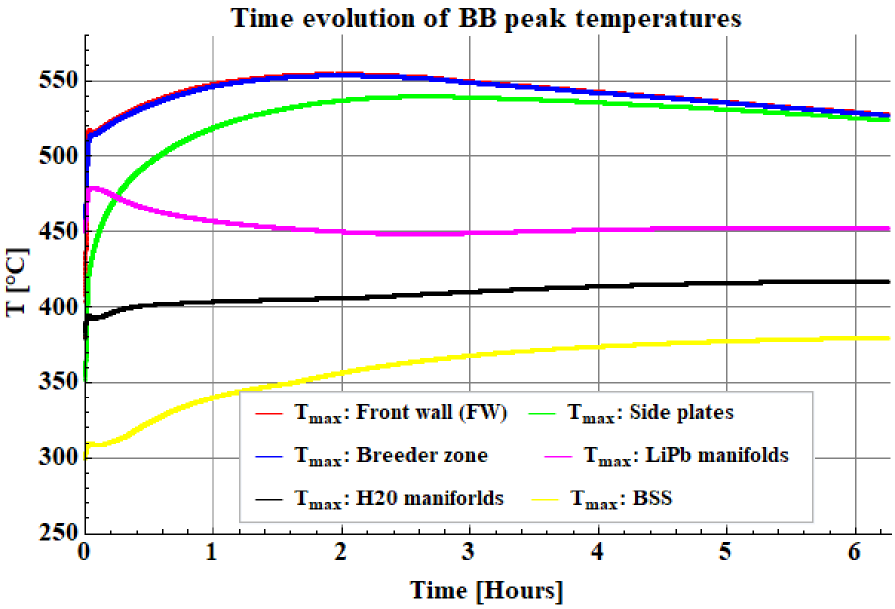


Figure 3: Time evolution of peak BB temperatures for central outboard segment.

Contours of temperature at external surfaces of the central outboard segment are shown in Figure 4. It may be observed that the highest temperature in BB occurs at the plasma facing side of the segment, i.e. at the front wall (FW).

Based on the distribution of thermal loads at external surfaces of BB segments, reported in [11], the ”effectiveness” of natural convection cooling vs. total removed “heat” is estimated as well as the estimation is made how the cooling is distributed between individual external surfaces. The results of analysis for central outboard (OutC) segment, obtained 1 hour and 6 hours after the LOCA occurrence, are reported in Table 3.

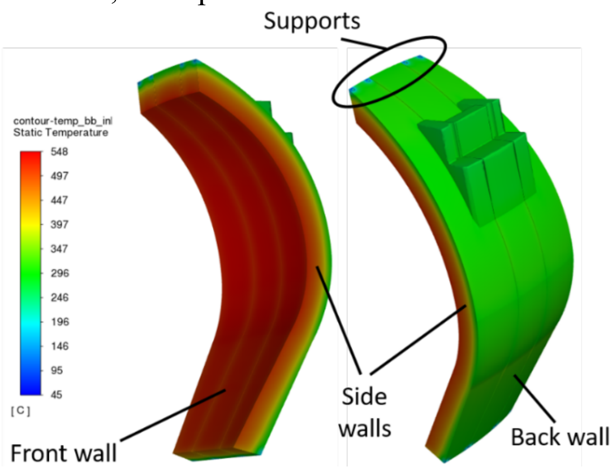


Figure 4: Temperature distribution on external surfaces of outboard segments. Cold spots are observed at locations of supports. Data are obtained 1 hour after the LOCA occurrence.

The percentages in the first four columns of Table 3 indicate that the natural convection cooling is the weakest at surfaces facing the centre of the plasma chamber (i.e. FW), where less than 40% of the removed heat per unit time through individual surface occurs due to natural convection. Instead, on surfaces facing the cold VV wall the contribution of natural convection cooling is substantially higher (~ 70%).

The last five columns of Table 3 report how the achieved cooling is shared amongst the individual external walls of the central outboard segment. It is observed that approximately 50% of the removed heat per unit time occurs through the back wall of the BB segment, while 20% (or less) of the cooling is achieved through the front wall.

Table 3: Distribution of thermal loads at external walls of central outboard segment, obtained 1 and 6 hours after LOCA occurrence.

	Contribution of natural convection cooling at individual external wall*				**Fraction of cooling at individual external surface with respect to total removed heat from segment [%]**				
Time	**@ FW [%]**	**@ side walls [%]**	**@ back walls [%]**	**@ chimney [%]**	**FW**	**side**	**back**	**Supp.**	**Chimney**
1 hour	22	60	67	71	18	9	51	2	20
6 hours	35	60	65	76	20	10	51	2	17

* Contribution of natural convection cooling = (natural conv.) / (natural conv. + radiation)

4 CONCLUSIONS

A three-dimensional transient conjugate heat transfer model has been developed to investigate the temperature evolution of DEMO in-vessel components (IVCs) during the ex-vessel LOCA for the EU DEMO BB concept. The analysis considers the Water-Cooled Lithium-Lead (WCLL) BB design concept [12]. The guillotine rupture of a large coolant pipe of the primary heat transfer system outside the vacuum vessel is assumed, resulting in instant failure of active cooling for all BB segments in all tokamak sectors. Hence, all heat is removed from the reactor by the safety-classified VV active cooling system. After plasma termination the VV is filled with helium to enable convective heat transfer of the decay heat from the uncooled BB to the VV. A full CFD transient analysis represents the most accurate approach to evaluate the transient temperature response of the in-vessel structures as it incorporates the convective cooling by the natural convection of filled gas and thermal radiation.

The CFD analysis shows that natural convection by filled helium seems to provide a sufficient cooling mechanism for decay heat removal from BB segments. Peak BB temperatures in the central outboard segment occur approximately two hours after the onset of the LOCA, reaching approximately 550°C.

Distributions of thermal loads at external walls of the central OB segment, estimated one and six hours after the LOCA occurrence, show that the natural convection cooling is the most effective on those sides of the BB segment facing the VV where approximately 70% of the heat is removed due to natural convection. On the BB FW instead, the heat transfer is dominated by radiation, while natural convection contributes only for ~40% or less. It is also observed that approximately 50% of total removed heat per unit time from the BB segment is provided at the back wall of BB segment, while the cooling at the front wall of BB segment is in order of 20%.

ACKNOWLEDGMENTS

This work has been carried out within the framework of the EUROfusion Consortium, funded by the European Union via the Euratom Research and Training Programme (Grant

Agreement No. 101052200 - EUROfusion). Views and opinions expressed are however those of the author(s) only and do not necessarily reflect those of the European Union or the European Commission. Neither the European Union nor the European Commission can be held responsible for them.

The financial support from the Slovenian Research Agency, grant P2-0405, is also gratefully acknowledged.

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