In-Vessel Loss of coolant accident assessment of a generalized LLCB TBS in ITER using RELAP/SCDAPSIM Code

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Abstract : *This work attempts to investigate the effect of In-Vessel loss of coolant accident of First Wall Helium Cooling System (FWHCS), of Lead-Lithium cooled Ceramic Breeder (LLCB) Test Blanket Module (TBM) system for ITER safety. The analysis discusses a number of safety concerns and issues that may result from the TBM component failure, such as VV pressurization, TBM FW temperature profile, passive decay heat removal capability and Suppression Tank (ST) pressure control capability. The analysis shows that in these accident scenarios the critical parameters have reasonable safety margins.*

Key words

ITER, Safety analysis, Thermal hydraulics, Test blanket system, RELAP/SCDAPSIM

1. INTRODUCTION

There is always crisis of energy around the world and in future energy requirement will be much more. Conventional sources of energy are not capable to fulfill the demand and are also dangerous for the environment. Non-conventional energy resources are an alternative. ITER ("The Way" in Latin) is one of the most ambitious energy projects in the world today. ITER is a fusion based research reactor which uses tritium as a fuel and will provide a large amount of energy. The TBM design will be tested in ITER to demonstrate the feasibility of its LLCB blanket concept and to access the TBM performance [1]. In-vessel TBM coolant leak case is analyzed to address ITER VV pressurization, VVPSS pressure control capability and passive decay heat removal capacity of TBM structure. The thermal hydraulic code RELAP5 is used to perform the analysis.

1.1 System description

The First Wall Helium Cooling System (FWHCS) transports the heat from the FW and the outer box structure. The TBM first wall is cooled by high pressure primary helium, which rejects heat to ITER water cooling system. The FWHCS is designed to remove the peak heat load of 300 kW [2]. The block diagram of FWHCS of LLCB TBM is documented in reference [3]. The complete thermal-hydraulic nodalisation diagram of FWHCS system is shown in Figure 1. The TBM FW composed of a 28 mm thick U-shaped RAFMS structure, having internal cooling channels of 20 mm × 20 mm cross section. The coolant channels are designed to allow multiple passes of helium coolant across the FW in order to maximize the heat removal. The number of helium passes has been optimized such that the maximum temperature in the RAFMS remains below the design limit of 550 °C. The FW structure is having 64 helium coolant channels [4]. The Vacuum Vessel Pressure suppression system (VVPSS) assembly consists of a large tank (suppression tank (ST)) of 46 m length and a circular cross section of 6 m diameter, volume of VVPSS is approximately 1,200 m^3 containing approximately 675.5m^3 of room temperature water at very low pressure (4.2kPa) to condense the steam coming from VV in case of severe in-vessel coolant leak accident. The suppression tank of VVPSS is located at level + 22.33 m above the VV. The Plasma Chamber (PC) of VV is connected through the D/HNB and HNB ports with DNBI and two D/HNBI module volumes that are connected by the relief pipes and relief manifold with the ST. There is a set of two consecutive rupture disks in relief pipe. The first rupture disk opens at a pressure difference of 150kPa and the second consecutive rupture disk opens at a pressure difference of 20kPa between the volumes separated by rupture disks. The design with double disk set is needed to avoid the influence of counter pressure downstream of the rupture disk due to the flow through the bleed line. There are two bleed lines with active valves attached to the relief pipe upstream and downstream of the rupture disk set. The active valves open if the pressure inside the PC is larger than 90kPa [3]. These bleed lines bypass the rupture disks in the relief pipe. The VVPSS assembly is shown in Figure 2 [4].

1.2 RELAP/SCADAPSIM code description

The RELAP/SCADAPSIM thermal-hydraulic code uses two-fluid one dimensional, non-equilibrium, non-homogeneous two-phase flow model to simulate the thermal-hydraulic characteristics of nuclear reactors. The code uses two-fluid model that can contain non-condensable components in the vapor/gas phase. This model consists of six governing equations to describe the mass, energy, and momentum of the two fluids and one equation for non-condensable. These group of equations are solved for seven primary dependent variables (pressure (P), phasic specific internal energies (Ug, Uf), vapor volume fraction or void fraction (αg), phasic velocities (vg, vf) and non-condensable quality (Xn)). The secondary dependent variables used in the equations are phasic densities (ρg, ρf), phasic temperatures (Tg, Tf), saturation temperature (Ts), and non-condensable mass fraction in noncondensable gas phase (Xni) for the i-th non-condensable species. Thermal hydraulic code RELAP/SCADAPSIM is a highly generic code used to calculate the behavior of a reactor coolant system during a transient. The code is also capable of simulation of a wide variety of thermal hydraulic transients in both nuclear and non-nuclear systems involving mixtures of steam, water, non-condensable, and solute. RELAP/SCADAPSIM is the latest version in RELAP5 series released by Innovative Systems Software (ISS).

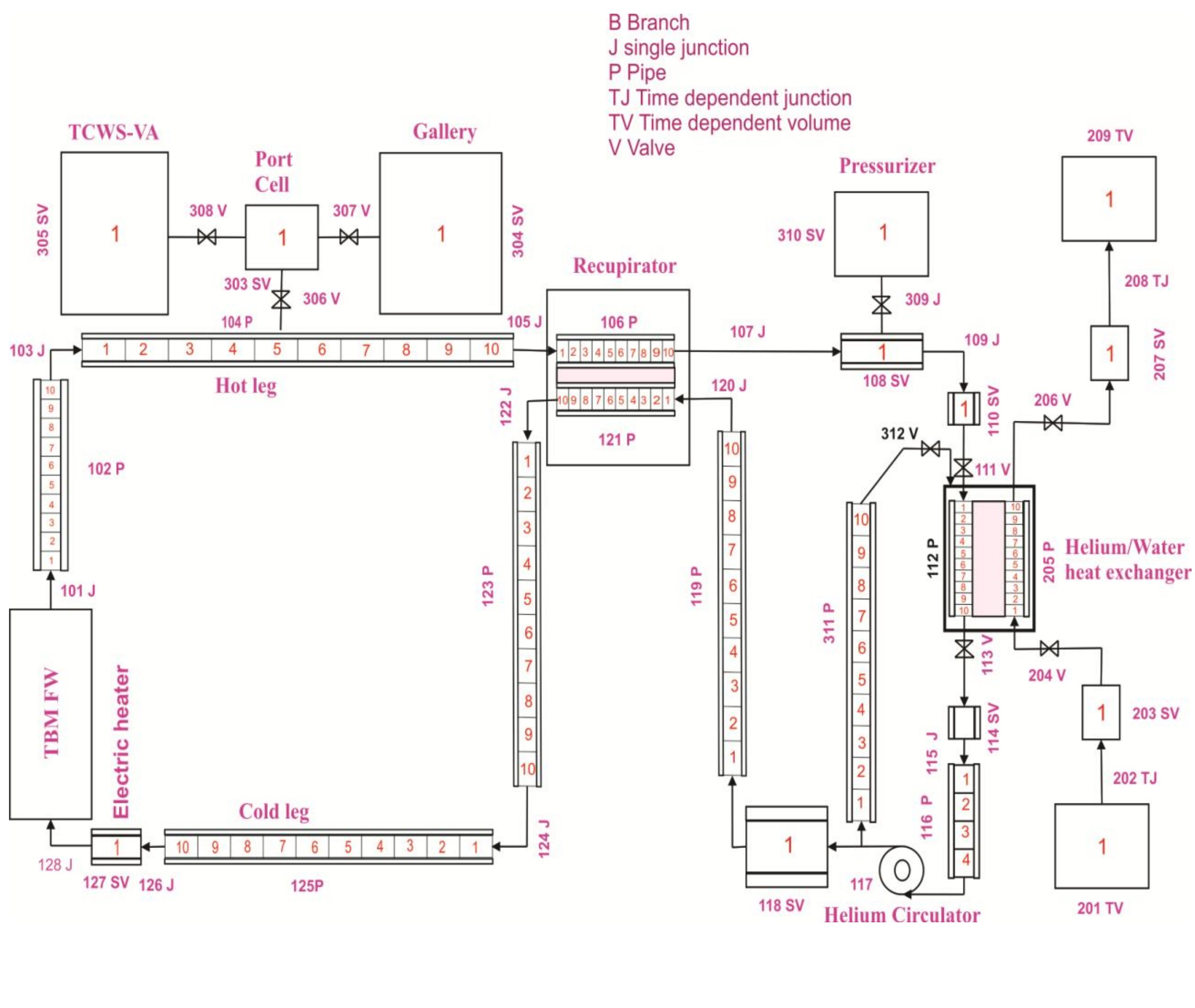

Fig.1. Thermal hydraulic nodalization of TBM FWHCS

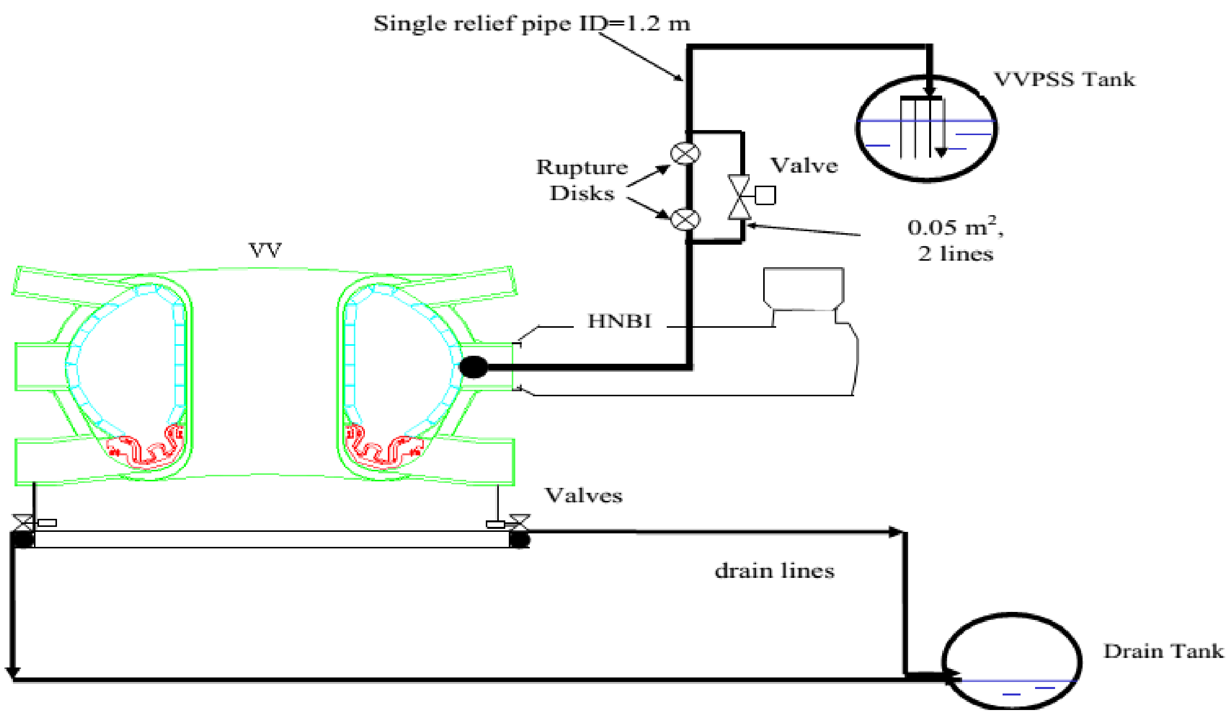

Figure 2 ITER VV and VVPSS assembly.

2. Method of Analysis

The analysis of the system is done with RELAP5 MOD3.4 code. The code uses Non-homogeneous, non-equilibrium two-fluid model for hydrodynamics, it uses the 1-D heat conduction model to define the structures with convective and radiative heat transfer capability [5]. The modeling of the TBM - FW cooling system is already discussed with flow diagram and RELAP5 nodalization in [2]. The cooling system of ITER-FW is modeled with time dependent volume with necessary system thermal hydraulics conditions along with a time dependent junction and a trip, representing the ITER-FW break. The hydrodynamic volume of VVPSS-ST is modeled by pipe component (component 326) having four volumes the total volume of the component is taken as 1200m^3, fluid for the first three volumes is selected as non-condensable (represents volume above pool) and water for the fourth volume (675.5m^3 represents pool volume). The assembly of VV consists of a PC (component 301), DV/DP (component 305) and NBI (component 310) and modeled with Single-Volume component with respect volumes and initial conditions. All the volumes are interconnected by either single-junction or valve component. The First Wall Helium Cooling System (FWHCS) is connected to the VV assembly by a Trip- Valve (component 304 in Figure 3) with flow area 0.0032m^2 represents a double ended break of four TBM-FW channels. The diverter volume DV/DP connected to the Drain Tank with a valve of flow area 0.0157 m^2 representing the drain pipe flow area, the trip is set to be activated after 1 hour of the accident. The complete interconnection of volumes is shown in nodalization diagram given in Figure 3. All the hydrodynamic volumes are connected with 1-dimensional heat-structures incorporate the thermal inertia of the structure.

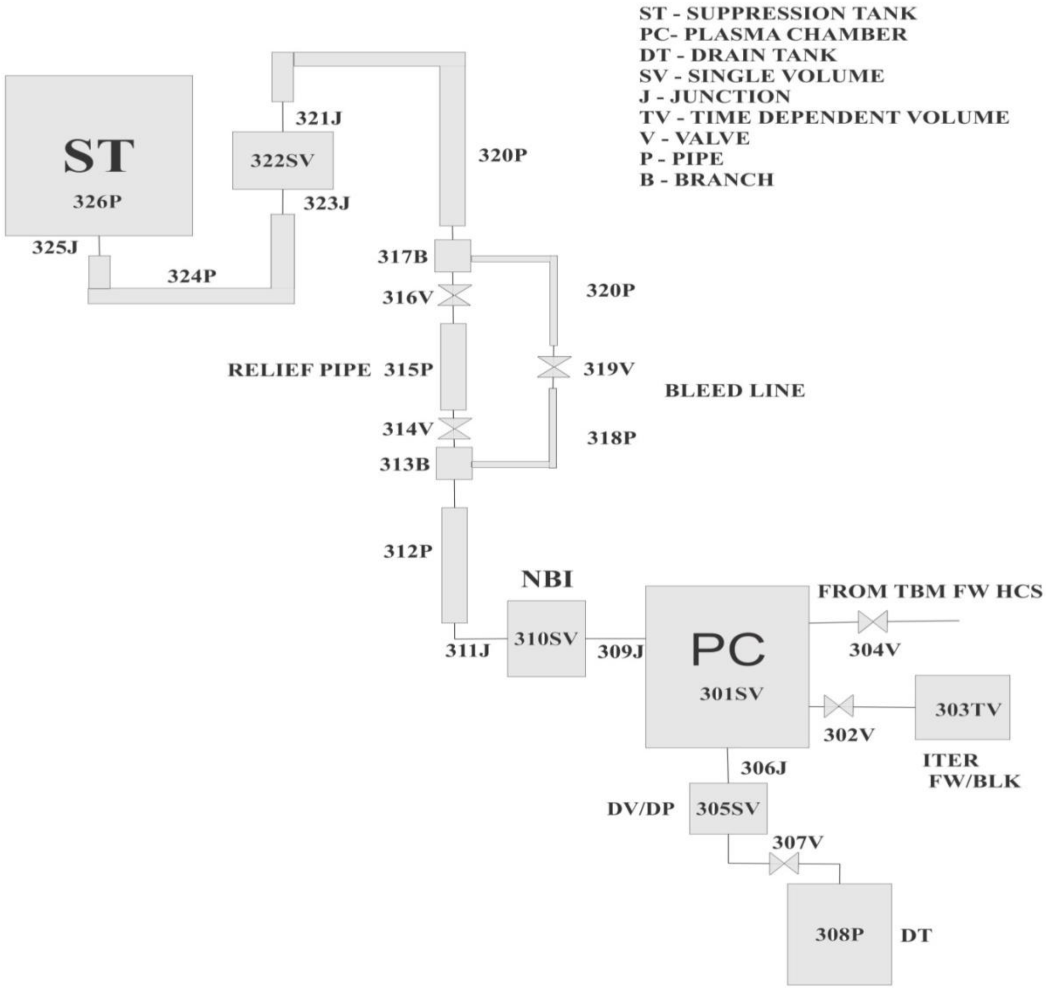

Figure 3 RELAP5 Nodalization diagram of VVPSS System.

3. Results and Analysis

3.1 Case 1: In-Vessel TBM Coolant Leak

3.1.1 Identification and Causes of accident:

This accident is considered as a design basis reference accident for the TBM. The postulated Initiating event (PIE) is the small break of the TBM-FW channel, results in a leak of TBM-FW coolant into ITER VV, prompt by TBM-FW weld failure. The ingress of helium into ITER Plasma induces intense plasma disruption and uniformly deposits 1.8 MJ/m2 of plasma stored thermal energy over FW within time assumed to be 1 s, generate runaway electrons which lead to the multiple TBM and ITER-FW cooling tube failures within a 10 cm high toroidal strip [6]. Water and steam blow down from the ITER-FW and Helium from the TBM-FW comes in to VV cause pressurization of VV. The size of the break has been defined as the double-ended rupture of all coolant channels within this toroidal strip around the entire reactor. This represents 4 FW channels (break size 0.0032 cm2) for the LLCB TBM. The pressurization causes the VV pressure suppression system (VVPSS) to open in an attempt to contain the pressure below the VV safety limit of 0.2MPa [3].

The transients due to TMB-FW helium ingress in VV (Plasma shut down by ingress of helium coolant (no ITER FW water coolant leak)) were discussed in [2], without any external active or passive pressure and temperature control system. The present analysis deals with transients and accidents due to TMB-FW helium ingress in VV along with the VV FW water coolant accident.

3.1.2 Results and Discussions

The steady state parameters obtained from the analysis [8], is used as the initial conditions for this accident scenario. The accident begins at the end of the flat top of a 500 MW pulse, or 500 s into the pulse. Maximum surface heat flux of 0.5 MW/m2 (0.3 MW/m2 normal) is given as input 10 s prior to the LOCA to guarantee peak TBM temperatures at the time of the accident. The break size of TBM FW is taken 0.0032 cm^2 representing 4 double ended ruptured TBM FW channels and ITER-FW cooling tubes break area is 0.02 m^2. The flow rate of water and steam is inserted by a table in the RELAP5 input deck [3]. Maximum surface heat flux of 0.5 MW/m2 (0.3 MW/m2 normal) is given as input 10 s prior to the LOCA to guarantee peak TBM temperatures at the time of the accident. A heat load of 1.8 MJ is given for 1 s after the LOCA as explained above. The helium ingress in VV causes plasma disruption and subsequent VV-FW failure causes the water at high enthalpy (622.44kJ/kg (3MPa and 148^{o}C)) blow down into the VV and flashes into steam. The mixture of helium and steam causes the pressurization of VV, at time t=9s after the accident as shown in Figure 4. The Vacuum Vessel pressure reaches 94kPa causes the bleed line open to capture the pressure of VV but the bleed line alone is not capable to control the pressure and it is further increase to 150kPa at t=17.5 sec, at this time both the rupture discs are break at this time both the relief line and bleed line are open and mixture from VV moves to ST rapidly where the steam is get condensed by mixing in low enthalpy water pool (125.7kJ/kg(30^{o}C and 4.2kPa)) and helium is collect in free space above the pool causes VV pressure arrested to 150kPa and further reduces at the same time the VVPSS pressure increases because of mixing of high temperature steam in pool and helium trap in VVPSS free volume causes reduction in mass flow rate of mixture comes from VV causes pressure slowly rise-up (65kPa to 75kPa) and after about 2000s of accident flow from the VV-FW cooling system reduces very low and further reduces to zero about 2800 Sec causes sharp decrease in VV pressure and settles to 47kPa. The drain tank valve is opened at 3600s results water goes through the drain line and collect in to drain tank chamber. This event is simulated for 10000secs in order to check VVPSS pressure venting and VV&TBS structure passive decay heat removal capability. The pressure profile of VV, VVPSS pool and drain tank is shown in Figure 4, the partial pressures of steam and helium are also shown in the Figure. The pressure profile of TBM-FW is shown in Figure 5, the pressure decreases sharply from TBM-FW operating pressure 8MPa and meets VV pressure at t=4sec as shown in the Figure. In this accident case; as mentioned earlier, a maximum surface heat flux of 0.5 MW/m2 was given for 10 s prior to the LOCA (t= 490 Sec in Figure 6). The temperature starts rising from 10 Sec before the accident from 475^{0}C to 530^{0}C and a peak temperature of ~645^{0}C is observed for 1 Sec at (t = 501s) in the TBM FW facing the plasma due to the deposition of heat load (1.8 MJ for a second) [8] and then the temperature reduces due to comparably low plasma shut after heat and cooling caused by the Water and Helium ingress into the VV. The complete TBM-FW temperature profile is shown in Figure 6. After a few seconds of the accident the FW cooling is completely shut down and after that heat balance is by radiation loss and conduction to colder structure. The temperature again starts increasing slowly after one hour of the accident t as shown in temperature profile, this is because initially the mass flow rate to the VVPSS reduces because of increase in VVPSS pressure and reduction in condensation in ST, the temperature again peaks (390^{0}C) at t=8000s at this time radiation loss, balance the decay heat and then after temperature start decreasing and reduces below 350^{0}C after 4 days due to the effectiveness of radiation loss from the structure.

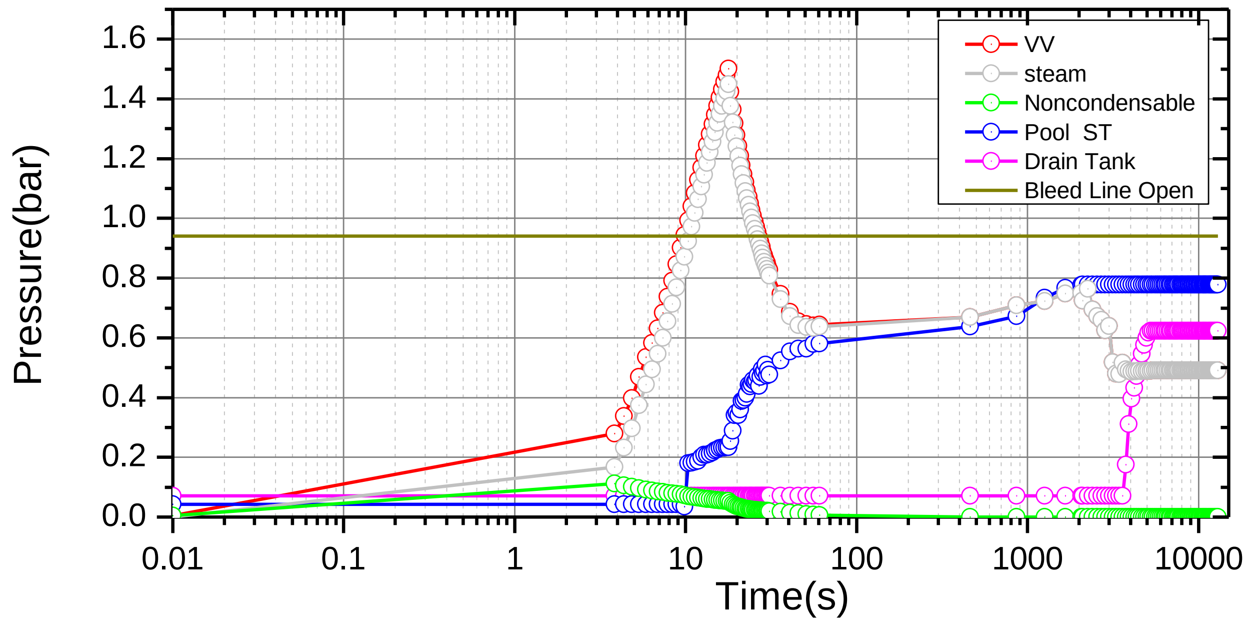

Figure 4 VV and VVPSS Pressure Profile In-Vessel TBM Coolant Leak case.

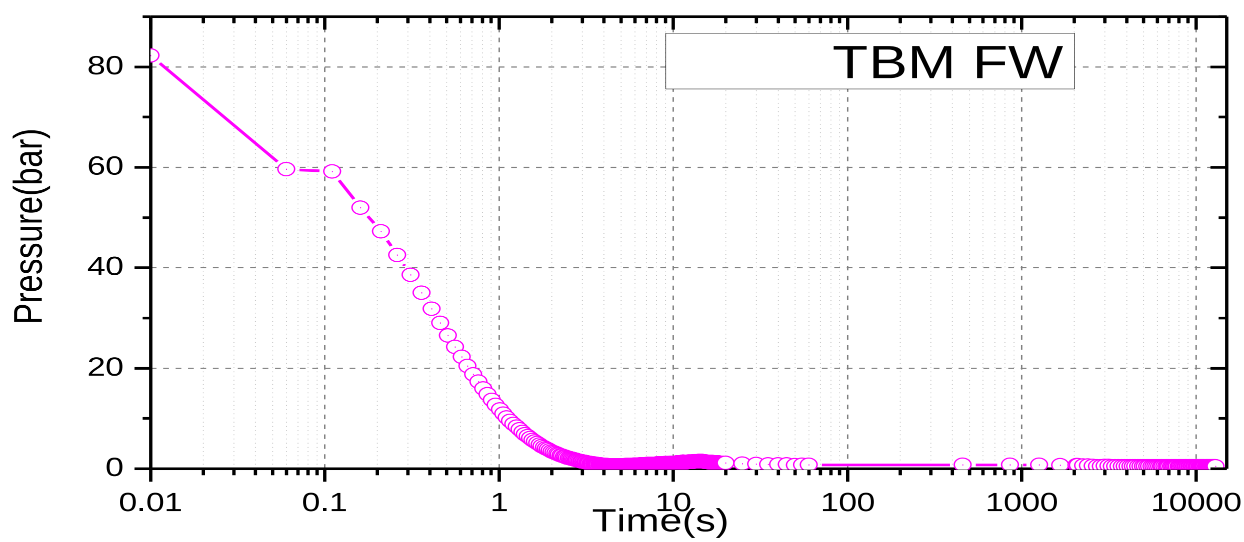

Figure 5 TBM-FW Pressure Profile In-Vessel TBM Coolant Leak case

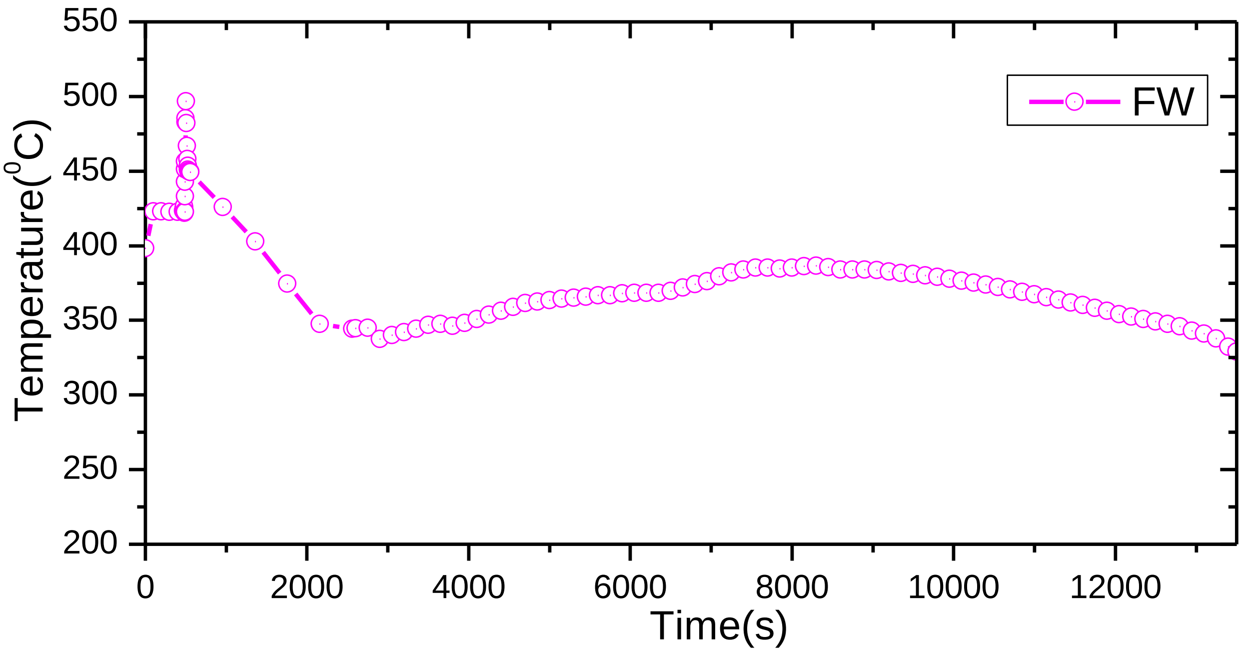

Figure 6 TBM-FW Temperature Profile

4. CONCLUSION

Analysis of the reference accidents: In-Vessel Helium leak with ITER-FW failure and ultimate safety related event: Ex-Vessel LOCA has been done. The analysis shows that the VVPSS capable to arrest the VV pressure under the design limit (200 kPa) set by ITER with a peak VV pressure of 150 kPa [7]. The temperature graphs show that the TBM-FW structure temperature gets a sharp peak and then keep on decreasing with time, indicates the structure is capable to remove decay heat passively.

5. SYMBOLS

FW	First Wall
FWHCS	First wall helium cooling system
FPSS	Fast Plasma Shutdown System
ITER	"The Way" in Latin
LOCA	Loss of Coolant Accident
LOFA	Loss of flow accident
PIE	Postulated initiating Event
ST	Suppression Tank
TBM/TBS	Test Blanket Module/Test Blanket System
VV	Vacuum Vessel
VVPSS	Vacuum Vessel Pressure Suppression System

6. REFERENCES

1. Saraswat, Satya Prakash, et al. Thermal Hydraulic Analysis of in-Vessel Loss of Coolant Accident and Loss of Flow Accident of First Wall Helium Cooling System of Generalized LLCB TBS in ITER Using Modified RELAP/SCDAPSIM MOD4. 0 Code. No. 2657. EasyChair, 2020.

2. S. P. Saraswat, P. Munshi, A. Khanna, C. Allison, (2017) " Ex-Vessel Loss of Coolant Accident Analysis of ITER Divertor Cooling System Using Modified RELAP/SCADAPSIM/Mod 4.0", ASME Journal of Nuclear Engineering and Radiation Science, Vol. 3, Issue 4, pp. 041009-1 to 041009-13.
    i. https://doi.org/10.1115/1.4037188

3. S. P. Saraswat, P. Munshi, A. Khanna, C. Allison, (2017) " Thermal Hydraulic and Safety Assessment of First Wall Helium Cooling System of a Generalized Test Blanket System in ITER Using RELAP/SCDAPSIM/MOD4.0 Code", ASME Journal of Nuclear Engineering and Radiation Science, Vol. 3, pp. 014503-1 to 014503-7.
    i. https://doi.org/10.1115/1.4034680.

4. S. P. Saraswat, P. Munshi, A. Khanna, C. Allison, (2018) "Thermal hydraulic safety assessment of LLCB Test Blanket System in ITER using modified RELAP/SCDAPSIM/MOD4.0 Code", ASME Journal of Nuclear Engineering and Radiation Science, Vol. 4 (2), pp. 021001 to 021001-10.
    i. https://doi.org/10.1115/1.4038823

5. S. P. Saraswat, D. Ray, P. Munshi, C. Allison, (2019) "Analysis of loss of heat sink for ITER divertor cooling system (new Tungsten divertor design) using modified RELAP/SCDAPSIM/MOD 4.0" , ASME Journal of Nuclear Engineering and Radiation Science, Vol. 5(4), pp. 042202-1-8.
    i. https://doi.org/10.1115/1.4042707

6. RELAP5/MOD3.3 Code manual Volume I, “Code Structure, System Models, And Solution Methods”, Nuclear Safety Analysis Division, Information Systems Laboratories, Inc., December (2001).

7. ITER, Generic Site Safety Report (GSSR) volume VII, “Analysis of reference events”, G 84 RI 6 01-07-10 R 1.0.