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Study of Decay Heat Removal and Structural Assurance by LBB Concept of Tokamak Components

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Abstract. Since decay heat density in ITER is quite low, thermal analyses have shown that only natural dissipation due to thermal radiation can be sufficient for removal of decay heat even in loss of all coolant. Owing to this attractiveness, no cooling system would be required for decay heat removal. In addition, because a magnetically confined plasma terminates by a small amount of impurity ingress, there is no possibility of uncontrolled production of energy, which will damage the integrity of the vacuum vessel containing tritium and other radioactive materials. This statement can be assured with a high level of confidence resulted from the LBB (Leak Before Break) concept.

1. Introduction

A tokamak type fusion device has inherent safety characteristics to shutdown plasma spontaneously at abnormal events. As in nature, fusion reaction has neither nuclear excursion nor chain reaction, and it is also self-bounded by physical limits such as pressure limit and density limit. Plasma also terminates by ingress of a small amount of impurities released from the surface of plasma facing components when over-heated due to loss of coolant, as well as a small water leakage through a through-wall micro crack penetration. In addition, residual heat density of in-vessel component due to activation is quite small (level of 0.3 MW/m^3 for ITER). Owing to these inherent characteristics, the escalation of abnormal event related to plasma and cooling systems to an accident is avoided without any safety provisions. With regard to safety assurance, therefore, the key principle is to assure integrity of containment barrier and to limit release of radioactive materials at postulated accidents within a safety limit. The inherent plasma shutdown due to a small water leakage can be the basis to assure the containment integrity by means of the LBB concept.

This paper presents analyses results on the inherent safety characteristics of decay heat removal using parameters of the new ITER (ITER-FEAT) and the experimental results on the applicability of the LBB concept.

2. Decay heat removal

In this study, 1D and 2D thermal analyses using the decay heat density of ITER-FEAT have been performed to calculate temperature transient of the tokamak components such as blanket, vacuum vessel, superconducting coils and cryostat.

2.2 1D thermal analysis

The analytical model simulates the tokamak outboard region composed of first wall (FW), shield blanket (SB), vacuum vessel (VV), thermal shield, toroidal field coil (TF coil) and the cryostat. The cryostat is considered as a final heat sink, whose temperature is fixed at 50˚ C because the cryostat is covered by radiation shield made by concrete with the thickness of 1 m [1]. Here, a new operation scenario (SA1) with maximum decay heat density is assumed. In this SA1 scenario, integrated fluence of 0.5 MWa/m^2 is added to the in-vessel component during 10 years and about one month of operation. In addition, the FW is conservatively assumed to be cooled by only radiation instead of conduction cooling.

Figure 1 represents the typical 1D analysis result under the worst conditions assuming loss of coolant in all cooling systems and only radiation cooling between adjacent components. Temperature of FW reaches the maximum of 660 ˚C at around 10^7 seconds (100 days) after the plasma termination. The temperature of VV increases from about 1 day and reaches around 620˚C at around 10^7 seconds.

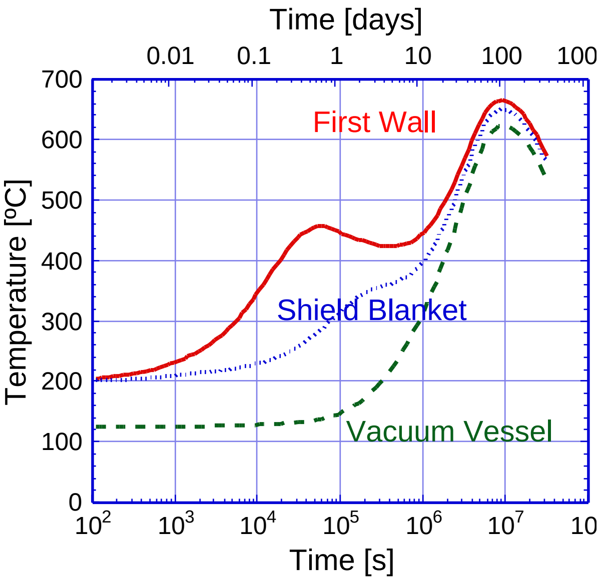

Fig.1 Temperature transient due to decay heat cooled by radiation

2.2 2D thermal analysis

2D thermal analysis has been performed in order to estimate maximum temperature of VV and FW including the temperature distribution effect. Thermohydraulic analysis code (MELCOR) is used for the analysis to take into account the effect of heat capacity of coolant in the cooling systems. For 2D model, poloidal structures of FW, SB and VV are separated to four poloidal parts, i.e. inboard, top, outboard and divertor. The inboard, top and outboard sides include the poloidal area of 6 blanket modules at the inboard side, of 3 blanket modules at the top and of 8 blanket modules at the outboard side, respectively. The VV structure simulates a double-walled configuration involving shield plate. A double wall is modeled as inner and outer shells. These shells are connected to the adjacent part of VV for the poloidal heat conduction. For inboard side, radiation heat exchange is taken into account to the outboard side. No heat exchange by radiation is considered toward top and divertor because of small view factor.

(1) Loss of coolant in all cooling systems

Figure 2 represents the temperature distribution of the radial direction at four time slices under the worst conditions assuming loss of coolant in all cooling systems. Initial temperature of TF and center solenoid coil (CS coil) are selected at 55 ºK as a temperature after the thermal quench of the superconducting coils. It has been shown that the temperatures of each structure are saturated about 680 ºC for FW and 620 ºC for VV at around t = 10^7 seconds,

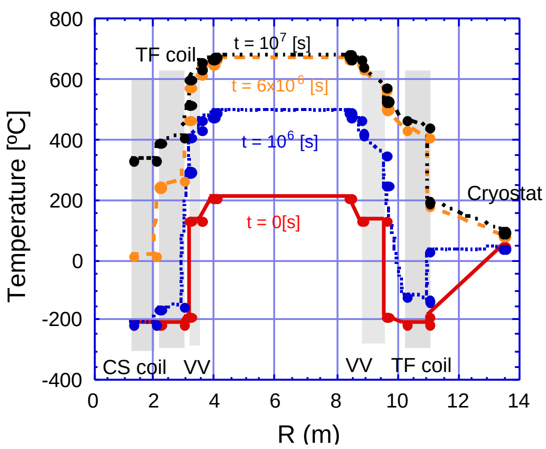

Fig. 2 Temperature distribution of the radial direction at 4 time slices

similar to those of 1D analysis results. The temperatures of the inboard TF and CS coil are still low compared with those of FW, SB and VV. It means that the TF and CS coils are expected to be heat sinks after 10^7 seconds. After that, the VV temperature becomes to reduce in time. Therefore, no more increase of the VV temperature can be expected.

On the other hand, the temperature at the outboard FW surface is slightly high compared with the temperature of inboard FW surface because the decay heat density is about one and a half times larger. It is because that the neutron flux density during operation on the outboard side is large owing to large view factor from the plasma. The temperature difference between inboard and outboard side is around 10 degrees. Consequently, radiation from outboard to inboard FW is small, in the order of 10 kW.

The VV temperature difference between inboard and top/divertor is also small. The conduction heat between them is an order of 10 W along the VV, which is negligibly small compared with the radiation between VV and SB in the of order of 100 kW.

(2) Effect of heat capacity of coolant in cooling systems

Effect of heat capacity of coolant in cooling loops has been assessed because loss of all coolant is highly improbable. In this analysis, three cooling systems are considered, i.e. FW/SB, divertor and VV cooling systems. Simple model of each cooling system is constructed by only a cooling loop, a pressurizer and a safety valve. No heat exchange from the cooling system is considered in this model.

Figure 3 shows the temperature transients of inboard side structures, coolant at SB outlet and saturation temperature of coolant determined by water pressure. Temperature transients of outboard and topside structures are almost the same since the cooling loop is connected. When the pressurizer is filled by water at about 3000 seconds, water pressure reaches the set point of the safety valve (6.4 MPa). After that, the coolant is released from the safety valve and the coolant inventory decreases. Boiling starts at 6×10^5 seconds when water temperature reaches the saturation temperature of 6.4 MPa. In this analysis, calculation stops at 1.5×10^6 seconds because the time steps become short more and more in time by frequent operation of the safety valve after the boiling. Temperature of VV becomes to increase at 10^5 seconds and reaches 190 ºC at 1.5×10^6 seconds, which is about 150 degrees lower than that in case of loss of coolant. Coolant inventory in FW/SB cooling system is about a half of initial inventory at 1.5×10^6 seconds. No water release from VV cooling system is observed. Therefore, maximum VV temperature can be expected to be at least 150 degrees lower than that without coolant.

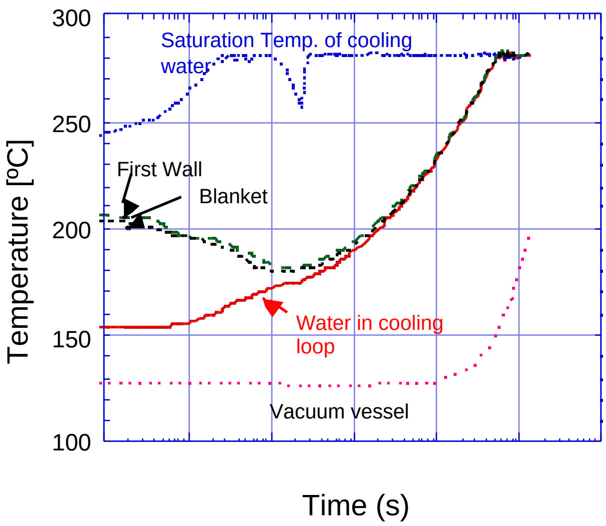

Fig.3 Temperature transient with water in cooling loops

3. Integrity assurance by LBB concept

The VV is a containment barrier of tritium and radioactive materials, and is made of SS316L in ITER that provides a leak before entire break (LBB) due to high ductility. This characteristic lead to be possible to check the leakage before the break of VV.

In addition, electromagnetic loads acting on VV are not uniform in the poloidal direction, resulting in local damage instead of entire structure damage. 3D FEM stress analysis has been performed by 1/40 of ITER VV supposing electromagnetic load during disruption. The maximum membrane stress of 138 MPa appears at the corner of the base of the support leg as a tensile stress. Therefore, extension of the crack can be possible by disruption load. The stress concentration factor at the tip of the crack has been estimated with a crack length normalized by material width of 1.77 m as shown in Fig. 4. The critical stress concentration factor is set to the 150 MPa•m$^{0.5}$, less than half of the fracture toughness as a conservative condition. For the criteria, a critical crack length for structural instability is more than 40 cm (distance from edge x/width ~0.22), which would cause a large amount of water leakage.

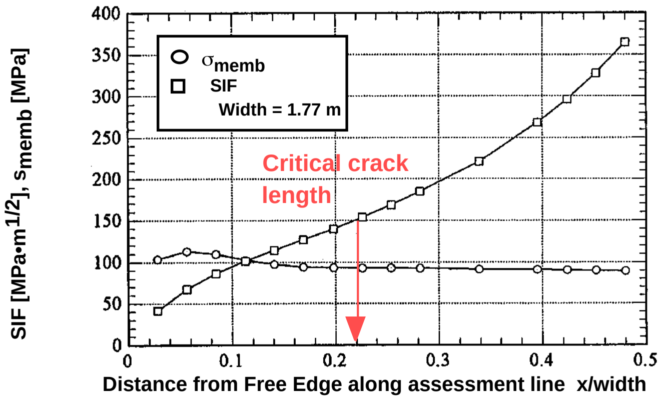

Fig. 4 Stress concentration factor (SIF) and the membrane stress with a distance from free edge

On the contrary, in the ITER in-vessel environment, a small water leakage to vacuum can be detected for increasing partial pressure of H_2O and O_2. The detectable partial pressure is around 10^{-7} Pa by mass-spectrometer. Using the effective pumping flow rate of 200 m^3/s, detectable leak rate is more than $2\text{x}10^{-5}$ Pa•m^3/s, around $2\text{x}10^{-8}$g/s for H_2O.

In order to verify the amount of water leakage through crack penetration to vacuum, the water leakage measurement has been performed as a function of crack size ranging from 1 mm to 5 mm. The crack was artificially formed to the test pieces that are made of type 316L stainless steel with a width of 50 mm and a thickness of 2 and 4 mm. The test piece was installed inside the vacuum chamber and pressurized up to 0.3 MPa by water. The water leakage through the crack was measured by a quadra-pole mass spectrometer connected to the vacuum chamber.

Table 1 summarizes the result of the leak rate with various test pieces. Crack lengths at front and back correspond to the crack length on the water-side and the vacuum-side of test piece surfaces, respectively. The results have shown that 1 mm to 5 mm crack at the vacuum side causes a water leakage of more than 10^{-8}g/s.

TABLE.1 CRACK SIZES AND THE LEAK RATE OF EACH TEST PIECES

Test piece No.	Crack length at front [mm]	Crack length at back [mm]	Leak rate [g/s]
2-1	7.75	5.00	$2.5\text{x}10^{-5}$
2-3	4.60	2.65	$7\text{x}10^{-6}$
2-6	2.33	0.78	$9\text{x}10^{-7}$
2-8	4.14	1.90	$1.5\text{x}10^{-5}$
4-1	4.30	3.15	$7\text{x}10^{-8}$
4-3	6.73	3.41	$6\text{x}10^{-8}$

In addition, an analysis model was developed for the estimation of water leakage. Straight tube with an elliptical cross-section was modeled as a leak path. The assumptions of this model are (1) surface inside the leak path is smooth, (2) interfacial surface exists inside the leak path and the pressure is balanced between liquid and vapor phases including surface tension, (3) temperature of wall, water and vapor are the same. When water-side pressure is lower than that of vapor side plus surface tension, vapor leak rate is evaluated as a vapor pressure of water surface. Considering the surface roughness and the undulation, locking parameter is used to fit the experiment, defined by effective gap of crack divided by measured or analyzed gap on the water-side surface of the test piece. Since the crack surface has some roughness and the undulation, a gap inside crack will be narrow in some place compared with the surface gap. The leak rate will be restricted by a conductance at the narrowest gap. The narrowest gap is estimated by assuming a geometry of crack surface refereed in ref.1. From the obtained narrowest gap from the estimation, the locking parameter is evaluated to be 0.14 for the test piece No.2-6. When the locking parameter is selected at 0.1~0.2, analysis results using this model has a best fit to the experimental results.

Using this model, leak rate of ITER VV has been estimated under the condition of the vessel thickness of 40 mm, a water pressure of 1 MPa and a water temperature of 120 ºC. Additional stress of 100 MPa was applied to the crack to simulate a disruption load. The crack length of the water-side was selected at 65 mm and the penetrated half-crack length was scanned. Analysis result shows that the leak rate was more than 10^{-8} g/s at the penetrated half crack length of 0.1 mm, as shown in Fig. 5, which can be detectable in the ITER high vacuum environment. When the penetrated half of crack length exceeds 0.4 mm, leak rate increases rapidly. Even in this crack length, the break of VV is not expected by the rapid extension of the crack.

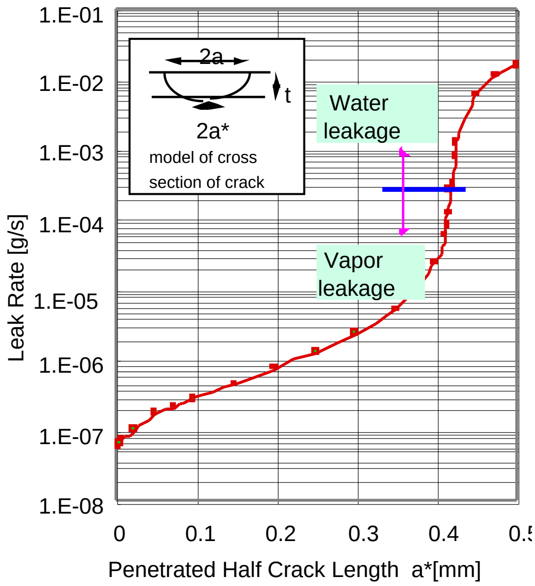

Fig. 5 Leak rate of ITER VV

4. Conclusion

In summary, after the operation, temperature of the VV is expected below around 600 ºC for decay heat even if no water is expected in the cooling loops. It can be concluded that any active cooling measures for decay heat removal is not required owing to small decay heat and large heat capacity of structures of ITER-FEAT as safety system, resulting that accommodates the safety requirements on occurrence of accidents that cause excessive release of radioactive materials due to melting of vacuum vessel. In reality, heat capacity of water in the cooling loops can be expected to reduce the VV temperature by more than 150 degrees.

The VV provides a leak before entire break (LBB) due to high ductility. 3D FEM analyses have shown that a critical crack length for structural instability is more than 40 cm. On the other hand, the experimental results have shown that 1 mm to 5 mm crack causes a water leakage of more than 10^{-8}g/s, which can be detectable in the ITER high vacuum environment. Therefore, it can be concluded that structural integrity of the ITER vacuum vessel can be assured by monitoring water leakage.

References

[1] TSURU, D., et al., submitted to 21st Symposium on Fusion Technology, Madrid, Spain, 10 – 15 September, 2000

[2] NAKABAYASHI, T., et al., Nucl. Engng. Des. 128 (1991) 17.