21pE-3

Study of loss-of-coolant events in a water-cooled tokamak DEMO
水冷却トカマク原型炉の冷却材喪失事象の研究

Makoto Nakamura, Kenji Tobita, Yoji Someya, Hisashi Tanigawa, Yoshiteru Sakamoto and [1]Werner Gulden
中村誠, 飛田健次, 染谷洋二, 谷川尚, 坂本宜照, [1]Werner Gulden

Japan Atomic Energy Agency, Rokkasho, Aomori, 039-3212, Japan
[1]Fusion for Energy, Garching D-85748, Germany
日本原子力研究開発機構　〒039-3212 青森県上北郡六ケ所村大字尾駮字表舘2-166
[1]Fusion for Energy, Garching D-85748, Germany

Thermohydraulic analysis has been performed for major in-vessel (in-VV) and ex-vessel (ex-VV) loss-of-coolant accidents (LOCAs) of a water-cooled tokamak DEMO by using the integrated thermohydraulic analysis code MELCOR with modifications for fusion reactor safety applications. The analysis identified thermohydraulic transients to these major LOCAs and the loads to the confinement barriers of radioactive materials. The ex-VV LOCA has been found to be quite challenging to integrity of the confinement vault of the water-cooled DEMO. The VV pressure suppression system of the present design has been found to be incapable of coping with the major in-VV LOCA.

1. Introduction

Safety characteristics should be demonstrated in a fusion DEMO reactor. Of late in Japan, a design study has been undertaken of a tokamak fusion DEMO with pressurized water coolant and solid pebble bed breeding blanket. The result of the preliminary hazard analysis [1,2] shows that the decay heat and the coolant enthalpy of a water-cooled DEMO, which can challenge integrity of confinement barriers, are significantly larger than those of ITER.A purpose of this study is to clarify accident propagations and possible loads to radioactive material confinement barriers of water-cooled DEMO from the viewpoint of thermohydraulics. We report thermohydraulics analysis of in-vessel (in-VV) and ex-vessel (ex-VV) loss-of-coolant accidents (LOCAs).

2. Modeling

A fusion DEMO reactor analyzed is accompanied with water-cooled pebble-bed blanket and LWR-like pressurized water coolant. Specifications are summarized in Table I and a schematic of the DEMO reactor is shown in Fig. 1.

Table I. Design parameters of the DEMO analyzed.

Parameter	Value
Major radius	8.3 m
Fusion power/	1.3–1.5 GW
Coolant	Pressurized water: 290–325 °C, 15.5 MPa
Armor mat.	Tungsten
BLK/DIV structural mat.	F82H
Tritium breed./ neutron mul. mat.	$Li_2TiO_3/Be_{12}Ti$ mixed pebble bed

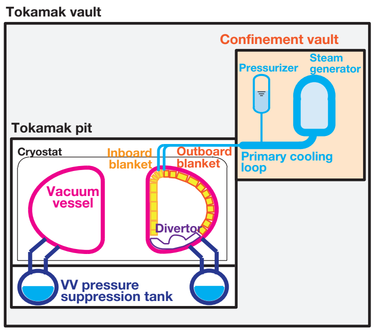

Fig.1. A schematic of the DEMO reactor analyzed.

We analyze in-VV and ex-VV LOCAs by using the integrated thermohydraulics analysis code MELCOR [3] with modifications for fusion reactor safety applications [4].

3. Ex-vessel LOCA

The initiating event is double-ended break of the cooling pipe outside the vacuum vessel but inside the tokamak building. Key input parameters are summarized in Table II.

The analysis results of the thermohydraulic transients to the ex-VV LOCA indicate that the following event sequences will happen following the ex-VV LOCA, as shown in Fig. 2. The analysis results indicates that (i) discharge of the liquid coolant water at the pipe break area lasts for 5.6 s after the pipe break and the coolant is completely lost and (ii) the pressure in the confinement vault containing the broken primary cooling loop, reaches the maximum of 144 kPa, which is ~40 times larger than the blowout panel set point of typical LWR building, at 97.6 after the pipe break. The results suggest that the ex-VV LOCA has been found to be quite challenging to integrity of the confinement vault of the water-cooled DEMO.

Table II. Key input parameters of the ex-VV LOCA analysis.

Parameter	Value
Coolant inventory	240 m^3/loop
# of the cooling loops	4
Inner diameter of the cooling pipe	0.727 m
Volume of the confinement area	380,000 m^3

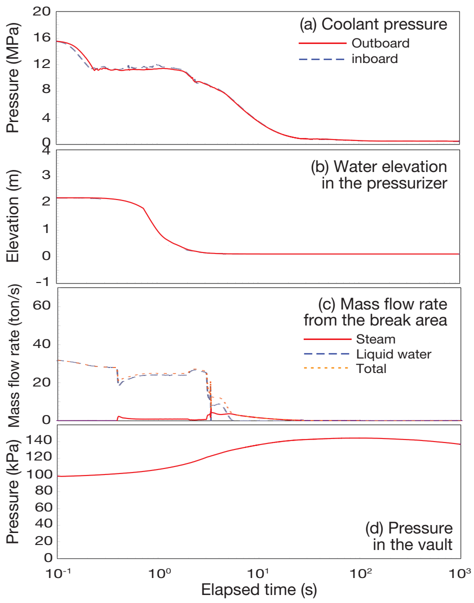

Fig. 2. Thermohydraulic transient behavior to the ex- VV LOCA.

4. In-vessel LOCA

The initiating event is multiple double-ended break of the FW cooling pipes. We conservatively assume break of the pipes of the whole outboard tokamak perimeter. It is also assumed that the FW cooling pipes are aligned in the poloidal direction and the total break area is 0.82 m^2 (6,430 pipes). Key input parameters are summarized in Table III.

Table III. Key input parameters of the ex-VV LOCA analysis.

Parameter	Value
Vol. of the VV	3,800 m^3
VV des. pressure	0.5 MPa
Vol. of the VV pressure suppression system (PSS)	5,600 m^3
Disk rupture set point between the VV and PSS	0.2 MPa
Area of the disk rupture	4.0 m^2
Cross section of a FW cooling pipe	64 mm^2

The analysis results of the thermohydraulic transients to the in-VV LOCA indicate that the following event sequences will happen following the in-VV LOCA, as shown in Fig. 3. The analysis result indicates that (i) the VV pressure reaches the rupture disk set point of 0.2 MPa at 0.61 s after the in-VV LOCA and (ii) nevertheless, the maximum VV pressure is 0.29 MPa at 6.5 s after the LOCA. The results sugest that the VV pressure suppression system of the present design has been found to be incapable of coping with the major in-VV LOCA.

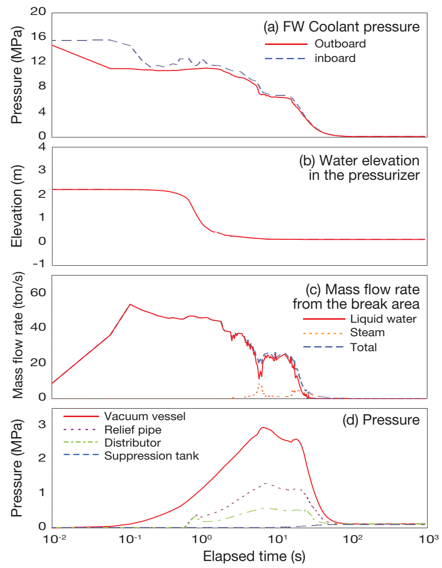

Fig. 3. Thermohydraulic transient behavior to the in- VV LOCA.

5. Concluding remarks

Thermohydraulic analysis of major in-vessel (in-VV) and ex-vessel (ex-VV) loss-of-coolant accidents (LOCAs) of a water-cooled tokamak DEMO identified the loads to the confinement barriers of radioactive materials. In the presentation, possible strategies are proposed to confine the radioactive materials and maintain the integrity of the confinement barrier will be discussed.

Acknowledgments

We thank Dr. Brad Merrill (Idaho National Laboratory, INL) for kindly providing us with MELCOR with the fusion modifications and his hospitality during their visit to INL. They also thank Mrs. Takao Araki (Toshiba Inc.) and Kazuhito Watanabe (Toshiba Inc.) for supporting MELCOR modeling.

References

[1] M. Nakamura, et al., Fusion Eng. Des., **89** (2014) 2028.

[2] M. Nakamura, et al., Plasma Fusion Res., **9** (2014) 1405139.

[3] R.O. Gauntt, et al., NUREG/CR-6119 (2000).

[4] B.J. Merrill, et al., Fusion Eng. Des. **51–52** (2000) 555.