Key Aspects of the Safety Study of a Water-Cooled Fusion DEMO Reactor*)

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Key aspects of the safety study of a water-cooled fusion DEMO reactor is reported. Safety requirements, dose target, DEMO plant model and confinement strategy of the safety study are briefly introduced. The internal hazard of a water-cooled DEMO, i.e. identification of hazardous inventories, identification of stored energies that can mobilize these hazardous inventories and identification of accident initiators and scenarios, are evaluated. It is pointed out that the enthalpy in the first wall/blanket cooling loops, the decay heat and the energy potentially released by the Be-steam chemical reaction are of special concern for the water-cooled DEMO. An ex-vessel loss-of-coolant accident (ex-VV LOCA) of the first wall/blanket cooling loop is also quantitatively analyzed. The integrity of the building against the ex-VV LOCA is discussed.

Keywords: water-cooled fusion DEMO, safety study, hazard analysis, accident scenario analysis, safety system

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1. Introduction

Fusion reactor safety has been considered since the beginning of fusion reactor design studies. (For more details on a historical overview of fusion reactor safety studies from the dawning to the end of 1980's, see Ref. [1].) A series of Safety and Environmental Assessment of Fusion Power (SEAFP) studies was conducted in Europe in 1990's [2, 3]. The conclusions suggest that safety characteristics of fusion reactors vary from reactor to reactor and that if a combination of materials or an output power is changed, safety characteristics of a fusion reactor will be altered. In parallel with and after the SEAFP studies, safety features particularly from the radiological viewpoint were demonstrated for various tokamak reactor designs, e.g. International Thermonuclear Experimental Reactor (ITER) [4, 5], Power Plant Conceptual Study (PPCS) [6, 7], ARIES-AT [8] and so on.

Safety characteristics should be demonstrated *not only* in ITER and future commercial fusion plants *but also* in a fusion DEMO reactor which is an intermediate step between ITER and a future commercial fusion plant. We study, for the first time, safety of a tokamak DEMO which is cooled by pressurized water, and of which blanket consists of solid pebble beds of tritium breeding material and neutron multiplying material. It is noted that the safety of a fusion reactor with such a combination of materials has not been studied extensively yet. The goals of this safety study [9, 10] are (i) to identify safety characteristics of a tokamak fusion DEMO reactor with such a combination of materials and (ii) to design and optimize safety-class and safety-significant structures, systems, and components.

In this paper, we report some aspects of the safety study of a water-cooled DEMO, in particular safety requirements, DEMO technical specifications, consideration on internal hazard of a DEMO with water-cooled pebble-bed blankets, some results of accident scenario analyses. This paper is an extension of the discussion in [11]. Besides the materials reported in [11], particularly in this paper (i) detailed discussions about the energies and power stored in the DEMO as internal hazard, and (ii) the results of the quantitative analysis of the ex-vessel loss-of-

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coolant accident, are presented. In Sec. 2, as the background of the safety research, safety requirements, a dose target and a DEMO plant model are reported. Analysis results are given in Sec. 3 on internal hazard of the DEMO. In Sec. 4, some preliminary results of qualitative and quantitative analysis of accident scenarios. Concluding remarks are given in Sec. 5. Appendix gives details of qualitative, logical accident scenario analysis.

2. Basis of the DEMO Safety Study

The safety approach and requirements for DEMO was presented in the earlier paper [11]. Here key aspects are recalled as background to the following sections.

2.1 DEMO plant model

The DEMO plant model is based on DT fusion in a steady state tokamak. Basic guidelines for finding out a DEMO design point, which is an input for the safety study, are as follows;

1. DEMO goals
    - DEMO should demonstrate net electricity output. Here we define the target net electricity of more than 200 MW, which that of is comparable to the demonstration fast breeder reactor Monju [12].
    - DEMO should demonstrate tritium self-sufficiency.
2. Technologies assumptions
    - The first wall (FW), breeding blanket (BLK) modules and divertor (DIV) cassettes are cooled by pressurized water in the pressurized water reactor (PWR) conditions (15.5 MPa, 290 - 325 °C). The vacuum vessel (VV) is cooled by water with lower pressure and temperature (e.g. 4 - 5 MPa, ~150 °C) than FW, BLK and DIV.
    - The tritium breeding blanket is made up based on know-how of the Japanese ITER Test Blanket Module (TBM) activities [13] and the Broader Approach (BA) DEMO R&D [14]. The structural material is made of reduced activation ferritic martensitic steel, F82H. Solid pebble beds made of lithium-titanate (Li_2TiO_3) and of Be-Ti beryllide ($Be_{12}Ti$) [15] are used as tritium breeding and neutron multiplying material, respectively.
    - The superconducting magnet technology is founded on that used in ITER. Nb_3Sn is used as superconducting material and the design conditions are assumed to be similar to ITER.

Assumptions about DEMO plasma physics are still to be decided. Forseeable integrated DEMO plasma performances is being studied [16]. Here just two premises are made about the plasma physics design: In order to ensure reliability of the operation start-up phase, the plasma current is ramped-up only inductively; After reaching the plasma current flattop, the plasma is operated in a steady-state mode.

We have calculated a set of DEMO design parameters by the fusion reactor systems analysis code TPC [17,18] coupled with the superconducting coil design code SCONE [19]. We performed extensive scans in a lot of input parameters of TPC and obtained large sets of DEMO design parameters. Of these sets, we selected a set satisfying the above guidelines. The DEMO design parameters selected are summarized in Table 1 and its radial build is shown in Fig. 1.

The major radius obtained by the systems analysis is larger than those of the previous DEMO designs such as SlimCS [20] and Demo-CREST [21]. This is because in order to ramp the plasma current up in an inductive manner, the radius of the center solenoid is larger than these two designs. It is noted that the volumes of the reactor building, tokamak cooling water system area, turbine area and hot cell, and their layout are to be determined.

Table 1 Design parameters selected. These are inputs for the safety analysis described in the following sections.

Parameter	Values
Major radius	8.3 m
Aspect ratio	3.0
Fusion power	1.3 - 1.5 GW
Net electricity output	200 - 300 MW
Coolant for in-vessel components	Pressurized water (290 - 325 °C, 15.5 MPa)
First wall armor material	Tungsten
Blanket structural material	Reduced activation ferritic martensitic steel F82H
Tritium breeding material	Lithium-titanate Li_2TiO_3 (pebble beds)
Neutron multiplying material	Beryllide $Be_{12}Ti$ (pebble beds)
Divertor armor material	Tungsten
Divertor structural material	Reduced activation ferritic martensitic steel F82H

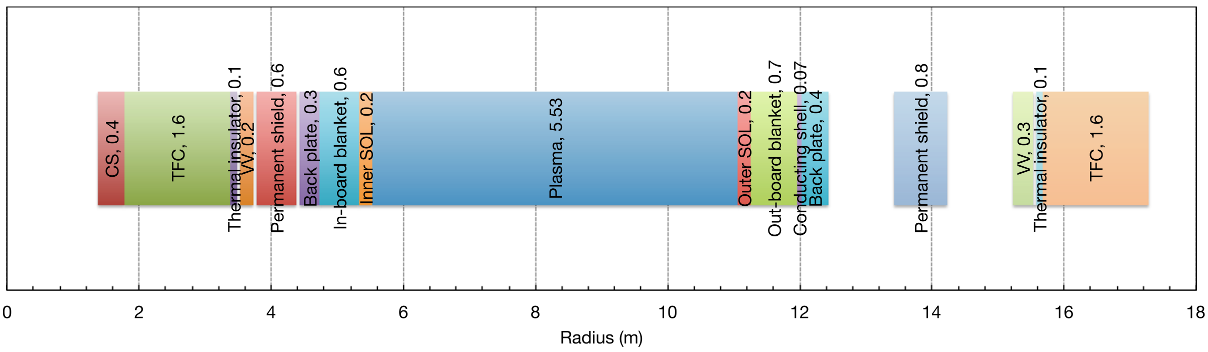

Fig. 1 Radial build of a reference DEMO used for the safety study. Numbers in the figure represent the thicknesses of the several parts.

2.2 DEMO safety requirements

Safety requirements adopted for the safety study are grounded on those shown in the Japanese ITER site invitation activities [22].

1. The public and fusion facility workers shall be protected appropriately against environmental release of radioactivity and release of radiation from the facility on normal conditions.
2. Accidents, i.e. release of radioactive materials from the facility, shall be prevented appropriately by ensuring the safety of systems containing the radioactive materials.
3. The impact of the accidents shall be mitigated so that environmental release of the radioactive materials is reduced and the off-site public dose is suppressed within a range in which the public is safe to the radiological risk.

2.3 Dose target

The dose target of the safety research has been decided to be 20 mSv. The dose target is ambitious compared with international dose guidelines, e.g. the evacuation-free criterion of < 50 mSv proposed by the International Atomic Energy Agency (IAEA) [23].

2.4 Confinement of the radioactive materials

The approach of confinement of the radioactive materials is based on ITER, i.e. it is assumed in this DEMO design that double confinement is implemented for in-vessel radioactive inventories. The first barrier is the vacuum vessel (VV), its extensions and cooling loops for the in-vessel components and VV; the second (final) barrier is the reactor building, cleanup systems and stack. While in some fusion reactor safety designs the cryostat is the second barrier, in the DEMO safety research presented here it is not. This is because the cooling loops, which are a part of the first barrier, will be connected to the heat exchanger outside the cryostat and inside the reactor building.

3. Hazard of the DEMO

Identifying hazards consist generally of three parts: (i) identification of hazardous inventories, (ii) identification of stored energies that can mobilize these hazardous inventories and/or break barriers confining these inventories, and (iii) identification of hazardous situations. In this section we report hazard analysis results along these three parts one by one. It is noted that analysis and discussions are restricted to internal radiological hazards of the tokamak and its ancillary systems of the DEMO. Non-radiological internal hazards, e.g. chemical toxicities of beryllium and nitrogen, electromagnetic risks, etc., and external hazards, e.g. earthquake, aircraft crash, external flooding, extreme climate conditions, etc., are out of scope of this paper.

3.1 Internal energies that can mobilize radioactive source terms

Stored energies which can mobilize radiologically hazardous inventories and/or break barriers confining these inventories in the tokamak building, include (1) plasma thermal and electromagnetic energies, (2) decay heat, (3) electromagnetic energies in superconducting magnets, (4) coolant energies and (5) chemical energies.

We evaluated these internal hazards. The energies stored in the plasma were calculated based on the DEMO design parameters evaluated by TPC. The magnetic energy of the superconducting coils was calculated using SCONE. The decay heat of the tokamak components were calculated by the neutronics/activation coupled calculation system developed in the Japan Atomic Energy Agency (JAEA) [24], in which the Monte-Carlo neutron transport calculation code MCNP-5 [25] is coupled with the activation calculation code THIDA-3 [26]. For estimation of the coolant energies the coolant inventory has to be given, but the primary cooling system has not been designed in detail yet. It was assumed that the volumes of water in the first wall/blanket (FW/BLK) and divertor (DIV) cooling channels and the number of the primary cooling loops are the same as those in SlimCS [27], i.e. the volumes and numbers of the primary cooling channels are 240 m^3/loop and 4 loops for the FW/BLK channel and 170 m^3/loop and 1

Table 2 Estimated internal energies as internal hazard of DEMO, that can mobilize radioactive materials.

Item	Value
Plasma	
Thermal energy (MJ)	870
Magnetic energy (MJ)	450
Magnet	
Magnetic energy (GJ)	120
Coolant	
Enthalpy in the first wall/blanket cooling channel (GJ)	1,300
Enthalpy in the divertor cooling channel (GJ)	230
Decay heat	
Just after shut-down (MW)	38
1 day after shut-down (MW)	8.3
1 month after shut-down (MW)	2.2
Chemical potential energy	
Beryllium (Be-steam reaction) (GJ)	31,000
Tungsten (W-steam reaction) (GJ)	200

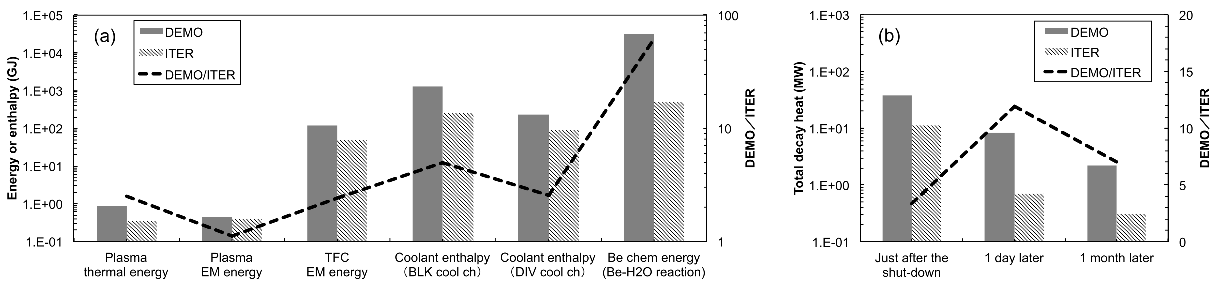

Fig. 2 Potential energy (a) and power (b) sources that can mobilize the radioactive materials expected in ITER and DEMO. The ratio of each source of DEMO to that of ITER is also shown. All the energy and power sources expected in DEMO are larger than those of ITER.

loop for the DIV channel, respectively. The coolant conditions are similar to PWRs, as indicated in the previous section. Hazardous chemical energy will be potentially released by the tungsten-steam reaction and beryllium-steam reaction. The former can occur in an in-vessel loss-of coolant accident (in-VV LOCA); the latter can occur in a LOCA inside a blanket module (in-BLK LOCA). (It is noted that beryllium is not used as plasma facing material in the water-cooled DEMO design.) Of these reactions the latter one could be more hazardous to the DEMO design because this reaction yields a bigger quantity of hydrogen, which causes an explosion hazard, than the W-steam reaction, and because the temperature excursion of the reaction has been observed [28]. It was assumed here that for calculating the chemical energy potentially released by the beryllium-steam reaction, the amounts of Be (760 ton) is similar to those in SlimCS [29].

Estimated internal energies of the DEMO are summarized in Table 2 and comparison with those of ITER shown in Fig. 2. These demonstrate that all of the energies and the decay heat involved in the DEMO are larger than those in ITER.

3.1.1 Energy of the coolant

The estimation result indicates that in the water-cooled DEMO, except for the chemical reaction potential energies, the enthalpy in the FW/BLK cooling loops is larger than the other energies involved in the DEMO. Also it is 5 times larger than that of ITER. This is because the coolant temperature is ~320 °C, which is larger than that of ITER (136 °C), for the purpose of electricity generation, and also because the coolant volume is larger than that of ITER (140 m^3/loop). This result suggests that a crucial issue is development of safety systems and confinement barrier strategies against the break of heat transport systems.

3.1.2 Decay heat

Also indicated in the estimation result is that the total decay heat in the DEMO is larger than in ITER in the wide range of time. (The decay heat expected in ITER is taken from [30].) One day later of the shut-down, the total decay heat of the DEMO is ~12 times larger than that of ITER. This is due to the decay heat of activation products

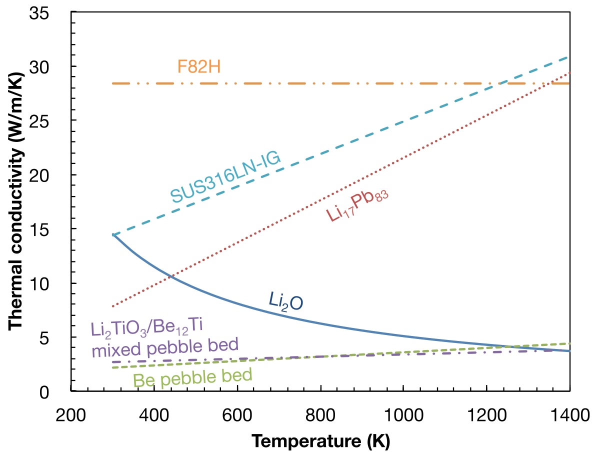

Fig. 3 Thermal conductivities of various materials expected to be used for fusion reactors: F82H [33, 34], SUS316LN-IG [35], $Li_{17}Pb_{83}$ [36], Li_2O [37], Be pebble bed [38] and $Li_2TiO_3/Be_{12}Ti$ mixed pebble bed [39]. The thermal conductivity of the $Li_2TiO_3/Be_{12}Ti$ mixed pebble bed is lower than other structural and functional materials for the blanket at the operation temperature range expected in the DEMO.

of tungsten of diverter modules of the DEMO, of which magnitude is dependent on the divertor configuration and on the neutron flux and fluence. It is noted that nuclear analysis of the full-W ITER diverter configuration is being re-assessed [31], and then the decay heat of the ITER divertor would be changed in the near future.

A strategy against an emergency situation of the DEMO might be needed for decay heat rejection because of such a large decay heat. In the ITER safety study [32] and the PPCS design [7], it was shown that under the extreme situation where all the coolants of the in-vessel components and vacuum vessel are completely and instantaneously lost, the temperature of each component of these tokamaks does not reach its melting point, because of radiative and conductive heat transfer toward the cryostat and natural air convection on the outer surface of the cryostat. Although the DEMO discussed here has the smaller fusion power and neutron wall load than those of PPCS series, it is unclear whether the DEMO has such a feature like ITER and PPCS series. This is because materials of the blanket and primary coolant of the DEMO are different from those of ITER and PPCS series. In fact, for example, the thermal conductivity of the lithium-titanate/beryllide mixed pebble bed is lower than other structural and functional materials for the blanket at the operation temperature range expected in the DEMO, as shown in Fig. 3. This intimates that the heat transfer characteristics of the DEMO are different from those of ITER and other reactor designs. Heat analysis of the DEMO under such a condition is addressed as a future work.

3.1.3 Energies potentially released by chemical reactions

Figure 2 and Table 2 indicate that the energy potentially released by the Be-steam chemical reaction is 1 - 4 orders of magnitude larger than the other internal energies of the DEMO, and is ~60 times larger than that expected in ITER. It should be mentioned that unlike the primary coolant and decay heat, this type of energy is indirect — it manifests only in the case where the coolant water is leaked inside a blanket module.

We consider that it is a crucial issue for the DEMO safety to suppress release of the Be-steam reaction energy and reaction-produced hydrogen in the vacuum vessel. Use of beryllide is a promising way for this. According to [40, 41], the reactivity of $Be_{12}Ti$ with steam is 2 - 3 orders of magnitude smaller than that of pure Be at the temperature of 1,000 °C, and the temperature excursion observed in the Be-steam reaction [28] has not been observed in the $Be_{12}Ti$-steam reaction. This is because the oxidized layer is formed on the surface of the $Be_{12}Ti$, which suppresses the propagation of the reaction and the thermal excursion [41]. An issue is that the $Be_{12}Ti$-steam reaction was observed only at the temperature of 1,000 °C. Further extensive experimental understanding of this reaction in the wide temperature range is needed for accident analysis of the DEMO safety study.

Table 2 also indicates that the energy potentially released by the W-steam chemical reaction is 2 orders of magnitude smaller than the Be-steam reaction. This is because the total amount of the W in the vacuum vessel is smaller than that of the Be. This result suggests that the W-steam reaction is less hazardous than the Be- steam reaction for the DEMO.

3.2 Radioactive source terms

Radioactive source terms expected to be involved in the DEMO are

1. tritium in the vacuum vessel
2. tritium permeated in the primary coolant
3. radioactive tungsten dust, originated from erosion of the first wall and diverter armors
4. activated corrosion products in the primary coolant.

These source terms are distributed mainly inside the first barrier.

In the safety study we have made an assumption on these source terms as summarized in Table 3, although there still remain uncertainties of these source terms.

3.3 Accident initiators and scenarios

We have analyzed accident scenarios and their initiators in a logical, qualitative way, i.e. by using the Functional Failure Modes and Effects Analysis (FFMEA) and the Master Logic Diagram (MLD), as was done for the Japanese ITER site invitation activities [42] and for ITER

Table 3 Assumption made on the radioactive source terms expected to be involved in the DEMO.

Source terms	Value
In the vacuum vessel	
Tritium	1 kg
Radioactive W dust	1,000 kg
In the primary coolant	
Tritium	TBD[a]
Activated corrosion products	TBD[a]

[a]to be determined

RPrS [5, 43]. The FFMEA deduced a lot of accident scenario branches that stems from an accident initiator. Of these branches we selected those deduced also by the MLD as accident scenarios which would potentially happen and for which some safety analyses or safety strategies are required.

The initiators of the accident scenarios selected are listed as follows;

1. Abnormal increase in the fusion power
2. Loss of coolant flow in the blanket cooling system
3. Loss of coolant flow in the divertor cooling system
4. Local increase in the heat load on the first wall
5. Transient thermal energy release due to the disruption
6. Loss of coolant flow after shut-down
7. Transient electromagnetic force due to the disruption
8. Quench of the toroidal magnetic field coils
9. Short of the toroidal magnetic field coil
10. In-vessel loss of coolant of the blanket cooling system
11. In-vessel loss of coolant of the divertor cooling system
12. Increase in the coolant pressure
13. Increase in the pressure of the He refrigerating system
14. Ingress of air at the cryostat boundary
15. Failure of cooling the isotope separation system
16. Failure of heater of the fuel storage bed
17. Break of the connecting port
18. Ex-vessel loss of coolant of the blanket cooling system
19. Ex-vessel loss of coolant of the divertor cooling system
20. Failure of the fueling line
21. Failure of the isotope separation system
22. Break of the cooling pipe in the breeding blanket

A full list of the accident scenario sequences resultant from these initiators are summarized in Appendix. These events are similar to those identified in the ITER safety study [42], except for the last one of the above list. Break of the cooling pipe in the breeding blanket, so called in-BLK LOCA, is a DEMO-specific event. For deterministic analysis for this event and resultant event sequence, e.g., the behavior of water spilled inside the blanket, heat transfer from the pebble beds to the water, blanket internal pressure and integrity of the blanket structure, it will be necessary to perform integrated thermo-hydraulic-structre simulation of the in-BLK LOCA.

4. Accident Scenario Analysis

In general, accident scenario sequences derived in a logical, qualitative manner are in turn analyzed quantitatively by using computer codes for design of safety systems and/or structures. Described here are the preliminary results of deterministic, quantitative accident scenario analysis of the ex-vessel (ex-VV) loss-of-coolant accident (LOCA) of the first wall/blanket primary cooling channel, which is one of the accidents derived based on the FFMEA and MLD analysis. It is remarked that the other accident scenario should be analyzed for safety design and assessment. We address this as a future work.

We have analyzed the ex-VV LOCA by using the fully integrated, engineering-level thermohydraulics analysis code MELCOR [44, 45] with modifications for fusion reactor safety applications [46, 47]. MELCOR is being used for several fusion reactor safety cases, for example ITER [47] and ARIES series [8, 48].

4.1 Assumptions and analysis conditions

We assumed that the ex-VV LOCA was caused by double-ended break of a primary cooling pipe outside the vacuum vessel but inside the tokamak building. The pipe break size is a key input of such an ex-VV LOCA analysis. We have not determined yet a design basis value of the break area. In the preliminary analysis reported here, we conservatively considered double-ended break of the cooling pipe. We also assumed that one of four primary cooling loops was broken. Key input parameters for the ex-VV LOCA analysis by MELCOR are summarized in Table 4. Components of the DEMO, i.e. the in- and outboard blankets, vacuum vessel, first wall/blanket primary cooling loops, heat exchanger and tokamak building, and the environment were nodalized by a small number of control volumes connected with mass and heat flow paths and with heat structures. A MELCOR model of this ex-VV LOCA is shown in Fig. 4.

4.2 Calculation results

The calculation result from the transient analysis of the internal pressure of the tokamak building is presented in Fig. 5. (In this paper, pressure is measured in absolute units unless there are special notations.) As indicated in the figure, the discharge of the primary coolant water pressurizes the building. Unless blowout panels are set up on the building wall, the internal pressure of the building reaches 144 kPa at 97.6 s following the ex-VV pipe break. Such increase in the pressure can be mitigated by setting up the blowout panels. In the case with the blowout pan-

Table 4 Key input parameters for MELCOR analysis of an ex-vessel loss-of coolant of a first wall/blanket cooling loop.

Parameter	Value
Volume of the vacuum vessel	3,800 m^3
Inlet coolant water temperature	290 °C
Outlet coolant water temperature	325 °C
Pressure of the coolant water	15.5 MPa
Coolant water inventory per a primary cooling loop	240 m^3/loop
Number of the primary cooling loops	4
Inner diameter of the broken cooling pipe	0.727 m
Break area	0.83 m^2
Volume of the building	380,000 m^3
Operation point of the blowout panels on the building	5.74 kPa (gauge)
Total area of the blowout panels	128.3 m^2

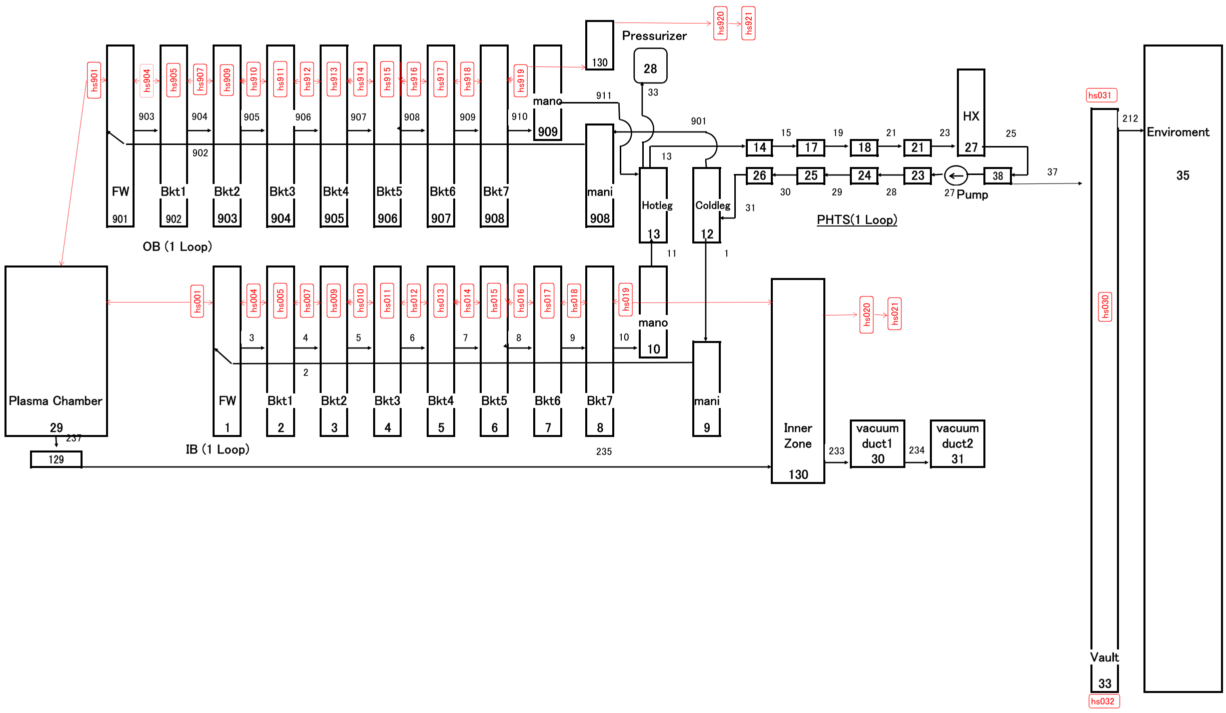

Fig. 4 Schematics of a MELCOR model of ex-VV LOCA. The mass and energy of the liquid water, vapor and other non-condensable gases in the in- and out-board blankets, vacuum vessel, first wall/blanket primary cooling loops, heat exchanger and tokamak building, and the environment are nodalized by a small number of control volumes. These volumes are connected with mass and heat flow paths. The heat flow between a volume and a component structure is also modeled by MELCOR. Such a structure is denoted by "hs" in this figure.

els, the building pressure reaches the operation point of the blowout panels, which open at 0.9 s following the pipe break, and in turn the heated air and tritiated steam are released to the environment. After that, the building pressure reaches the maximum of 113 kPa at 3.1 s and is decreased mainly because of the release of the heated air and tritiated steam.

The calculation result of the integrated mass of the tritiated steam released to the environment is presented in Fig. 6. As indicated in the figure, the integrated mass of the steam released to the environment reaches 16 tons at 135 s following the pipe break.

4.3 Discussion of the ex-VV LOCA calculation results

The analysis results suggest that the operation of the blowout panels has significant impact on the integrity of the building against the ex-VV LOCA. In the case without the blowout panels, it is unclear whether the building can withstand the pressurization, of which maximum is 144 kPa, caused by the ex-VV LOCA. Possible approaches to avoid break of the integrity of the building are (i) to en-

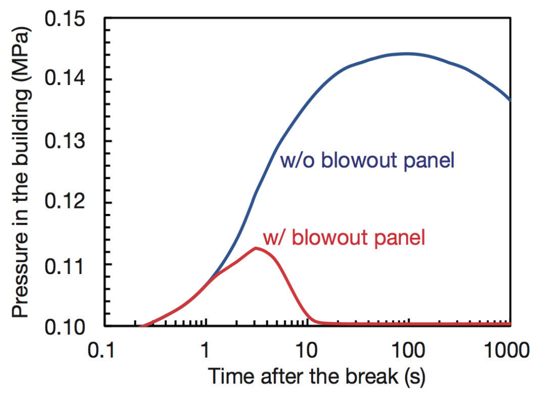

Fig. 5 Internal pressure of the tokamak building in the cases with and without the blowout panels on the building wall. In the cases without the blowout panels, the internal pressure of the building reaches 144 kPa (abs) following the ex-V pipe break. Such increase in the pressure can be mitigated by setting up the blowout panels.

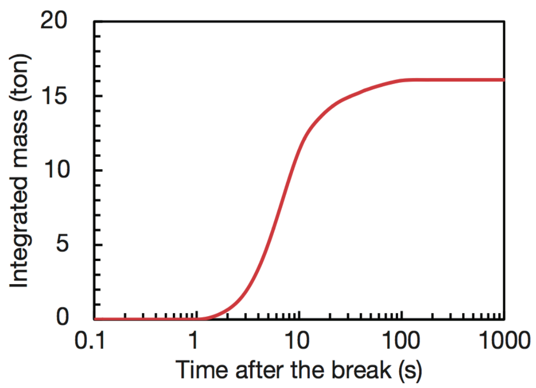

Fig. 6 Integrated mass of the tritiated steam released to the environment by the ex-VV double-ended pipe break. The calculation indicates that the total mass of the steam released is 16 tons.

large the volume of the building, (ii) to cover the primary cooling loop and heat exchanger by a primary heat transport vault, of which design pressure is larger than that of the building, and (iii) to install a pressure suppression system for the building or vault. Not only these approaches, as indicated in Fig. 5, but also to implement the blowout panels can mitigate pressurization to the building and maintain the integrity of the building. This approach will be employed if the amount of tritium and ACPs is so small that the dose to the public is below the dose target.

Consequences of the ex-VV LOCA, e.g., dose to the public at and outside the site boundary should be evaluated. Tritiated water and ACPs are contained in the released steam, and the dose to the public is dependent on such radioactive source terms. Permeation characteristics of tritium through F82H pipes and corrosion characteristics of F82H by the hot water of the DEMO-like operation condition should be analyzedsufficiently to the water-cooled DEMO safety design.

5. Summary

We reported some key aspects of the safety study of a water-cooled DEMO. After briefly introducing the safety requirements, dose target, DEMO plant model and confinement strategy, we presented assessment of the internal hazard of the water-cooled DEMO. It was shown that DEMO internal energies that can mobilize radioactive materials are different from those of ITER. In particular, the enthalpy in the FW/BLK cooling loops and the total decay heat in the DEMO one day later of the shut-down are 5 and 12 times larger, respectively, than those of ITER. The energy potentially released by the Be-steam chemical reaction is 1 - 4 orders of magnitude larger than the other internal energies of the DEMO, but it should be mentioned that this type of energy manifests only in the case where the coolant water is leaked inside a blanket module. Radioactive source terms likely in the DEMO was assessed. At the present stage, unfortunately, it is difficult to quantify precisely the inventories of tritium and ACPs in the primary coolant. The accident initiators and scenarios were analyzed by the FFMEA and MLD methods. We identified the 22 accident initiators followed by the accident sequences. We performed the deterministic, quantitative analysis of the ex-VV LOCA of the FW/BLK cooling loop, that is one of the initiators identified, by using MELCOR. Consequences of the accident, i.e. pressurization of the building, release of the water to the environment and so on, were assessed. The analysis results suggest that the operation of the blowout panels has significant impact on the integrity of the building against the ex-VV LOCA.

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Appendix. Accident Scenario Sequences in a Water-Cooled DEMO

Here listed are the accident initiators and accident sequences caused by the initiators derived based on the FFMEA and MLD analysis.

1. Abnormal increase in the fusion power
    - Increase in the temperature of the divertor plate
    - Plasma disruption due to break of divertor cooling channels

- Break of in-vessel components cooling pipes due to the disruption
- Ingress of water in the vacuum vessel
- Pressure suppression system in operation and condensation of steam water

2. Loss of coolant flow in the blanket cooling system
    - Heating of the first wall by the plasma
    - Plasma disruption due to break of a cooling channel of blanket or ingress of impurity into the plasma
    - Ingress of water in the vacuum vessel
    - Hydrogen production by the chemical reaction of the first wall material with steam water
    - Pressure suppression system in operation and condensation of steam water
3. Loss of coolant flow in the divertor cooling system
    - Heating the divertor plate by the plasma
    - Plasma disruption due to break of divertor cooling pipes or ingress of impurity into the plasma
    - Break of blanket cooling channel due to the disruption
    - Ingress of water in the vacuum vessel
    - Hydrogen production by the chemical reaction of the divertor material with steam water
    - Pressure suppression system in operation and condensation of steam water
4. Local increase in the heat load on the first wall
    - Local heating of the first wall (due to the plasma vertical displace event or failure of plasma heating systems)
    - Plasma disruption due to break of first wall cooling channel
    - Ingress of water in the vacuum vessel
    - Hydrogen production by the chemical reaction of the first wall material with steam water
    - Pressure suppression systems in operation and condensation of steam water
5. Transient thermal energy release due to the disruption
    - Break of the cooling channel of in-vessel component due to the disruption
    - Ingress of water in the vacuum vessel
    - Pressure suppression system in operation and condensation of steam water
6. Loss of coolant flow after shut-down
    (a) Break or melting of in-vessel component cooling channel
    (b) Ingress of water in the vacuum vessel
    (c) Pressure suppression system in operation and condensation of steam
7. Transient electromagnetic force due to the disruption
    - Break of in-vessel components cooling channel due to the disruption
    - Ingress of water in the vacuum vessel
    - Pressure suppression system in operation and condensation of steam water
8. Quench of the toroidal magnetic field coils
    - Decrease in the current of in the toroidal magnetic field coil
    - Plasma disruption due to the loss of the toroidal magnetic field
    - Break of in-vessel components cooling channel due to the disruption
    - Ingress of water in the vacuum vessel
    - Pressure suppression system in operation and condensation of steam water
9. Short of the toroidal magnetic field coil
    - Decrease in the current of in the toroidal magnetic field coil
    - Quench of the coil due to increase in the current in the shorted coil
    - Plasma disruption due to the loss of the toroidal magnetic field
    - Break of in-vessel component cooling channel due to the disruption
    - Ingress of water in the vacuum vessel
    - Pressure suppression system in operation and condensation of steam water
10. In-vessel loss of coolant of the blanket cooling system
    - Ingress of water in the vacuum vessel
    - Pressure suppression system in operation and condensation of steam water
11. In-vessel loss of coolant of the divertor cooling system
    - Pressure relief valves open for the coolant loop
12. Increase in the coolant pressure
    - Pressure relief valves open for the coolant loop
13. Increase in the pressure of the He refrigerating system
    - Pressure relief valve open of the He refrigerating system
14. Ingress of air at the cryostat boundary
    - Decompression in the building
    - Quench of the toroidal magnetic field coils
    - Decrease in the current of in the toroidal magnetic field coil

- Plasma disruption due to the loss of the toroidal magnetic field
- Break of in-vessel component cooling channel due to the disruption
- Ingress of water in the vacuum vessel
- Pressure suppression system in operation and condensation of steam water

15. Failure of cooling the isotope separation system
    - Increase in the hydrogen isotope gas pressure in the isotope separation system
    - Pressure suppression system for the isotope separation system in operation
16. Failure of heater of the fuel storage bed
    - Increase in the fuel gas pressure in the fuel storage bed
    - Pressure suppression system for the fuel storage system in operation
17. Break of the connecting port
    - Plasma disruption due to ingress of air into the vacuum vessel
    - Break of in-vessel components cooling pipes due to the disruption
    - Ingress of water in the vacuum vessel
    - Pressure suppression system in operation and condensation of steam water
    - Release of the steam in the vacuum vessel to the reactor building
    - Diffusion and absorption of the steam, tritium and dust in the building
18. Ex-vessel loss of coolant of the blanket cooling system
    - Increase in the gas pressure in the reactor building and release of radioactive materials from the stack or the blowout panel
    - Decrease in the coolant pressure in the cooling pipes
    - Heating of the first wall by plasma
    - Plasma disruption due to break of first wall cooling channel or ingress of impurity into the plasma
    - Break of in-vessel component cooling channel due to the disruption
    - Hydrogen production by the chemical reaction of the first wall material with steam water
    - Pressure suppression system in operation and condensation of steam water
    - Flow of tritium and activated dust from the vacuum vessel into the reactor building, through the empty pipe
19. Ex-vessel loss of coolant of the divertor cooling system
    - Increase in the gas pressure in the reactor building and release of radioactive materials through the stack or the blowout panel
    - Decrease in the coolant pressure in the cooling pipes
    - Heating of the divertor plate by plasma
    - Plasma disruption due to break of divertor cooling pipes or ingress of impurity into the plasma
    - Break of in-vessel component cooling channel due to the disruption
    - Hydrogen production by the chemical reaction of the first wall material with steam water
    - Pressure suppression system in operation and condensation of steam water
    - Flow of tritium and activated dust from the vacuum vessel into the reactor building, through the empty pipe
20. Failure of the fueling line
    - Ingress of the tritium gas to the tritium area
    - Exchange of the tritium with the air in the tritium area
    - Tritium burning in the fueling line
    - Removal of the tritium gas
21. Failure of the isotope separation system
    - Ingress of the tritium gas to the tritium area
    - Exchange of the tritium with the air in the tritium area
    - Tritium burning in the fueling line
    - Removal of the tritium gas
22. Break of the cooling pipe in the breeding blanket
    - Ingress of the steam in the coolant to the blanket module
    - Break of pipes of the tritium recovery system
    - Break of the blanket module
    - Pressure suppression system in operation and condensation of steam water
    - Release of radioactive materials through the broken pipes of the tritium recovery system
    - Break of the vacuum vessel and release of the radioactive materials (if the pressure suppression is failed due to non-condensable gas)

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