Concept of power core components of the SlimCS fusion DEMO reactor

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Abstract. This paper presents a novel concept of power core components of the SlimCS DEMO reactor with a focus on water cooling, including blanket, conducting shell and high temperature shield. Based on the design study, we propose feasible conceptual designs on breeding blanket and high temperature shield. The ideas will be commonly applicable to any other water-cooled reactor.

1. Introduction

JAEA has conducted the conceptual design of the SlimCS DEMO reactor (FIG.1) characterized by a compact and low aspect ratio (A = 2.6) concept with a major radius of 5.5 m, minor radius of 2.1 m, maximum field of 16.4 T, normalized beta (β_N) of 4.3 and fusion output of 2.95 GW [1]. For high β_N access, the reactor requires a conducting shell at a near distance from the plasma, e.g., $r_{shell}/a \leq 1.35$. This requirement can impede ensuring self-sufficient fuel supply because the blanket thickness is limited to about 0.5 m when a is as small as 2 m. Water cooling adopted in the reactor has advantages in efficient heat removal and overwhelmingly low pumping power over helium cooling. On the other hand, it may reduce tritium fuel production in the blanket. In addition, water cooling requires a challenge regarding engineering design for high pressure tightness.

This paper describes a way to deal with these conflicting requirements on water cooling and presents a feasible concept of water-cooled power core components.

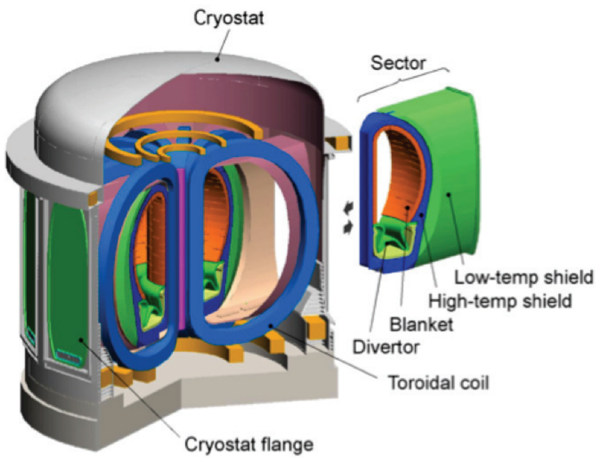

FIG.1 Conceptual view of SlimCS

2. Configuration of power core components

The blanket needs to produce tritium enough to meet self-sufficiency of fuel. For the purpose, each blanket module should be as large as possible so as to reduce non-breeding zones like a gap between the neighboring blanket modules, rims of the blanket, etc. To the contrary, the module size should be small as possible to withstand electromagnetic (EM) forces acting on disruption. In order to balance these competing requirements, the blanket modules of SlimCS are segmented into toroidally-long modules with the size of 1.4-2 m (toroidal) ×0.5-0.6 m (poloidal) in light of the EM force analysis [2]. In the original design, the modules were mounted on the high temperature shield (HT shield) with a single poloidal-ring structure (FIG.2 (a)) which provided adequate strength against turn-over forces on disruption.

Recently, the HT shield was replaced by the assembly of HT shield blocks as shown in FIG.2 (b). This is a consequence of considerations on how to contain high pressure water in the HT shield. The vacuum vessel of ITER has double wall structure and form a poloidal ring similar to FIG.2 (a), which is made by welding steel plates and is filled with steel and water inside the double wall for shielding [3]. Why it is possible to adopt the plate welding in ITER is that the vacuum vessel has no risk to burst due to the inner pressure of the double wall because the water for shielding is used at the temperature below 100ºC. That is, the double wall is a nonpressure vessel (inner pressure ~0.1 MPa) in ITER. In contrast, when water is used in the HT shield of DEMO, the water condition in the HT shield is basically the same as that in the blanket, being higher than ~300ºC and ~15 MPa. The double wall structure made by plate welding is no more applicable to contain such high pressure water, and instead, the water needs to be contained in high-pressure piping.

A possible concept of the HT shield is to form cooling channels by drilling forged steel blocks. Since a forged material with the thickness of > 0.6 m might show a degradation of strength due to coarse material texture, we determined the thickness of the shield blocks to be about 0.6 m in the poloidal direction. The blocks are made of reduced activation ferritic martensitic steel (F82H) [4]. Each shield block on the outboard side is inserted in a frame structure from behind as shown in FIG.2 (b). The frame structure is boarded with a conducting shell on the outboard side at the position of r_{shell}/a = 1.35 for plasma vertical stability and high β_N access. On the inboard side, the blanket modules attached with shield block are inserted in the frame from the inside.

The conducting shell is sector-wide and saddle-shaped. It has a fin at each end of the toroidal direction so that the assembly of the shells cancels unfavorable components of eddy current [5] and thus can improve the plasma stability and high β_N access. The conducting shell is made of the F82H plate with the thickness of about 7 cm. Since nuclear heating of the shell surface can be as high as 3 W/cm^3, the resulting heat needs to be removed by coolant. For the purpose, cooling tubes are embedded in the shell as shown in FIG.3, the cooling tubes are connected with the outlet header of the coolant of the HT shield.

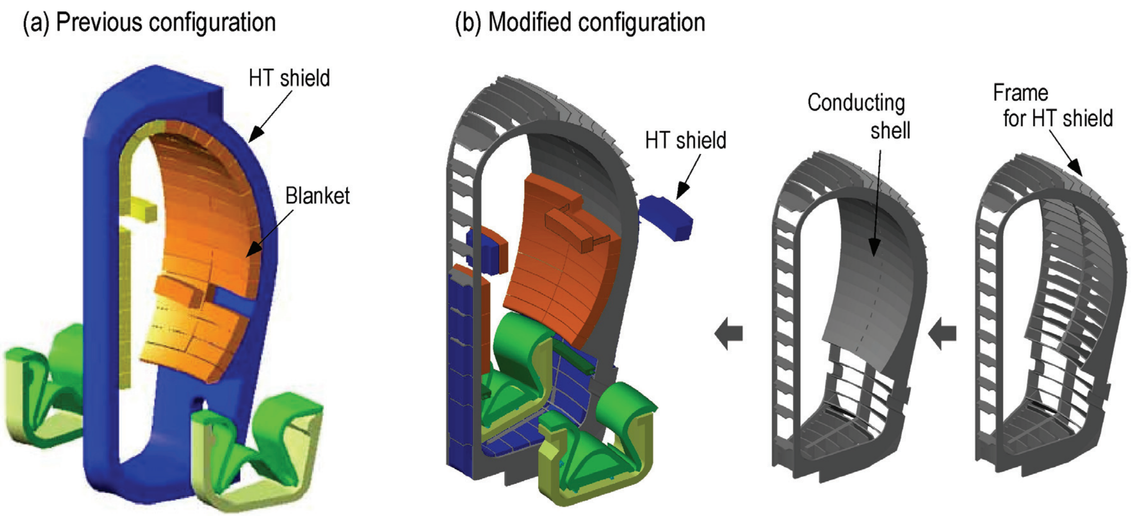

FIG.2 Assembly concepts of power core components: (a) previous concept of HT shield with poloidal ring structure and (b) modified concept composed with small HT shield blocks.

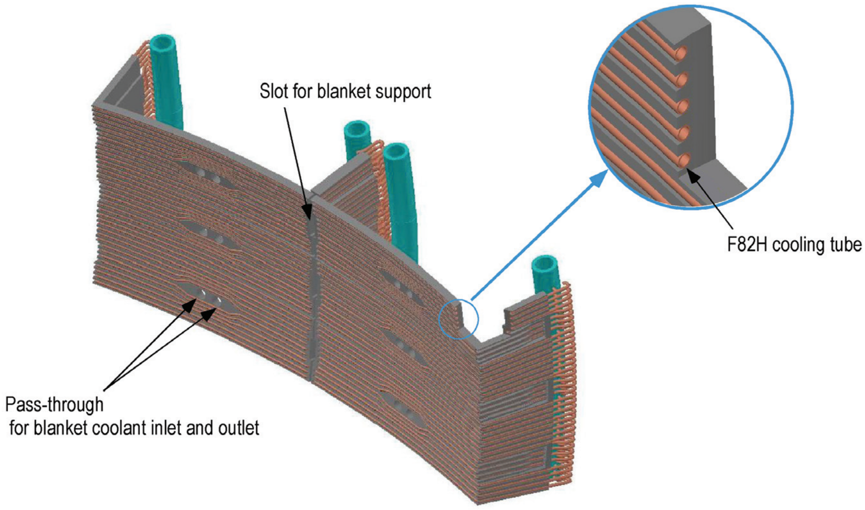

FIG.3 Conceptual view of the conducting shell

3. Solid breeder blanket with simplified structure

Considering the continuity with the Japanese ITER-TBM program [6] based on water-cooled solid breeder, blanket design on SlimCS originally focused on a "multi-layered structure" concept as proposed in Ref. [7]. The prime option for tritium breeder is Li_4SiO_4 since neutronic calculations indicate its superiority in TBR compared with Li_2TiO_3 and Li_2ZrO_3. This is mainly because, compared with the other lithium ceramics, Li_4SiO_4 has high Li density per unit volume. $Be_{12}Ti$ and Be are considered as neutron multiplier. Generally speaking, when Be is used as the neutron multiplier, a breeding blanket shows higher TBR due to absence of Ti. In spite of less TBR, chemical stability of $Be_{12}Ti$ is attractive in the viewpoint of system design in that it does not react with hot water in case of breaking of coolant boundary. In addition, use of $Be_{12}Ti$ allows a simple blanket structure because no tight separation from a breeding material is required because of a chemical stability of $Be_{12}Ti$. Considering these pros and cons of such materials, the blanket interior was originally designed to have a structure installing breeder and multiplier zones alternatively as shown in FIG.4(a). Notice that Be layers with the F82H casing are allocated in the front layers of the blanket to attain high TBR as possible.

Previously estimated tritium breeding ratio (TBR) for the multi-layered structure was about 1.05 when the blanket coverage was taken into account, indicating marginal self-sufficiency of fuel [8]. However, a heat analysis of cooling water in the blanket, which was carried out after the TBR calculation, indicated that the water condition at the outlet was about 370ºC of temperature and 6.5 m/s of flow speed. These values exceeded the water condition target set in the beginning of the design; that is, subcritical water condition with the temperature range of 290-360ºC and flow speed of < 6 m/s. In addition, the multi-layered structure seems to have engineering difficulties in 1) welding of cooling tubes and the F82H casing of Be layer, 2) gap control of each layer, 3) pebble packing in narrow regions and 4) fabrication time of such complex structure. In order to resolve the difficulties, an alternative concept with simplified structure is considered.

In the alternative concept, the blanket is filled with the mixture of chemically stable Li_4SiO_4 and $Be_{12}Ti$ pebbles and the F82H casing for Be is completely removed [9] as shown in FIG.4(b). The concept has a distinctive advantage of fabrication in that the component structure inside the blanket is cooling tubes and the support only. Furthermore, such a simplified structure has a possibility of increasing TBR due to a relative increase of the fraction of breeding materials. The concern is that replacement of Be by $Be_{12}Ti$ may lead to a reduction in TBR.

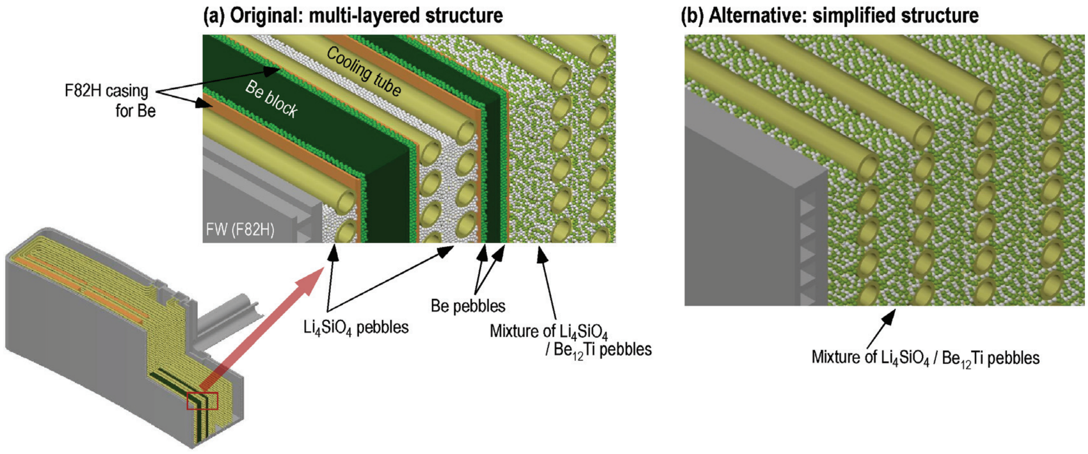

FIG.4 (a) Original blanket concept with multi-layered structure and (b) an alternative concept with simplified structure.

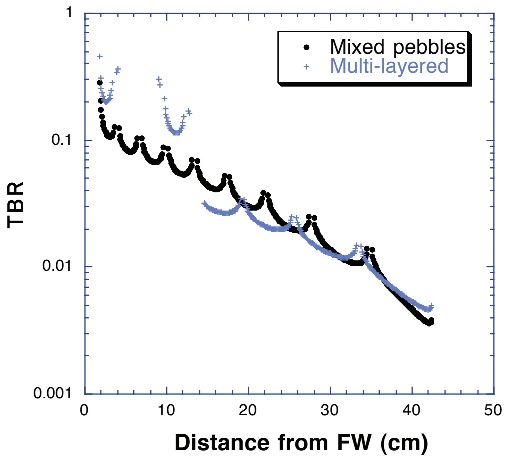

FIG.5 Comparison of local TBR as a function of the distance from the first wall.

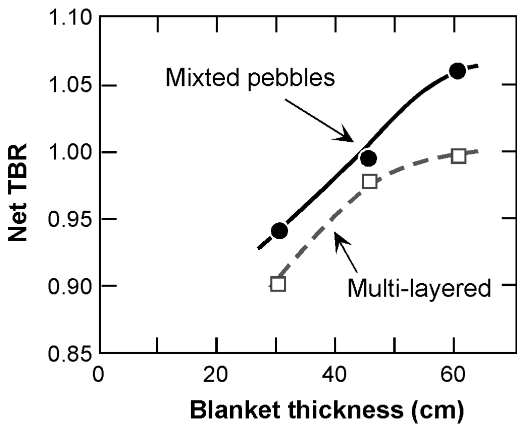

FIG.6 Net TBR for the mixed pebbles and the multi-layered blanket [8].

FIGURE 5 shows the spatial distribution of TBR for an optimal blanket design with the simplified structure when the neutron wall load (P_n) of 3 MW/m^2 in comparison with that for the original multi-layered design. In both cases, the water fraction in the blanket was given to satisfy the outlet water temperature of 360ºC and the flow speed of 6 m/s. Notice that the pebble mixture blanket steadily produces tritium throughout the blanket although the TBR in the forward layers is smaller than TBR of the multi-layered blanket. The total TBR for the mixed pebbles blanket obtained by integrating the TBR along the blanket depth, is higher. Considering the blanket coverage on the first wall, the net TBR was estimated for the mixed pebbles (Si_4LiO_4 and $Be_{12}Ti$) blanket and the multi-layered one with different blanket thickness as shown in FIG.6. The result indicates that the tritium breeding capability of the mixed pebbles blanket is superior to that of the multi-layered one. In addition, taking account of ease of fabrication, the mixed pebbles blanket should be regarded as a more promising water-cooled solid breeder blanket.

4. High temperature (HT) shield

In fusion power reactor concepts, neutron shield is usually divided into high temperature shield (HT shield) and low temperature one. The HT shield is located behind the blanket. Nuclear heating of the HT shield is as high as 1 W/cm^3 and the gross nuclear heating of the HT shield can reach a few hundreds of 100 MW. For efficient use of nuclear heat, therefore, the cooling channel of blanket should be connected with that of the HT shield in series. As a result, the temperature of the HT shield is maintained at a high temperature comparable as the blanket. When the shield is composed of F82H and water, the fraction for efficient shielding should be typically F82H : water = 0.7 : 0.3. The problem is how to contain a plenty of water of 290-360ºC, 23-25 MPa in the shield. Because of the difficulty of pressure-proof design, a feasible concept of water-cooled HT shield has not been proposed so far.

FIGURE 7 depicts a novel concept of the HT shield resolving the problem. Pressure-tight cooling channels for shielding are produced by drilling. Firstly, poloidal cooling channels are drilled in the front and back parts of the F82H block. Secondly, the poloidal cooling channels are connected in a series by R-directional cooling channels as shown in FIG.8 (a). Finally, each drilling entry is sealed by welding a plug. Notice that, since the HT shield is a permanent core component to be used throughout the life of reactor, R-directional channels should be drilled from the back of the F82H block to

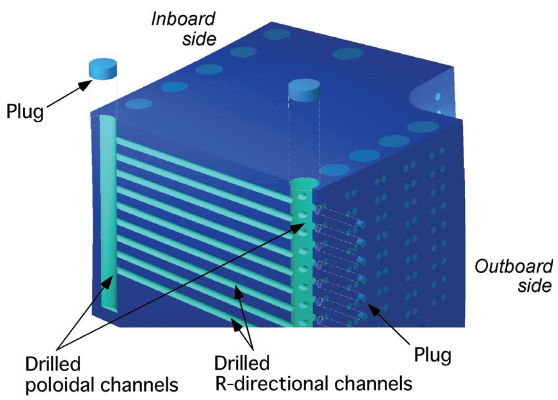

FIG.7 Concept of HT shield with cooling channels made by drilling.

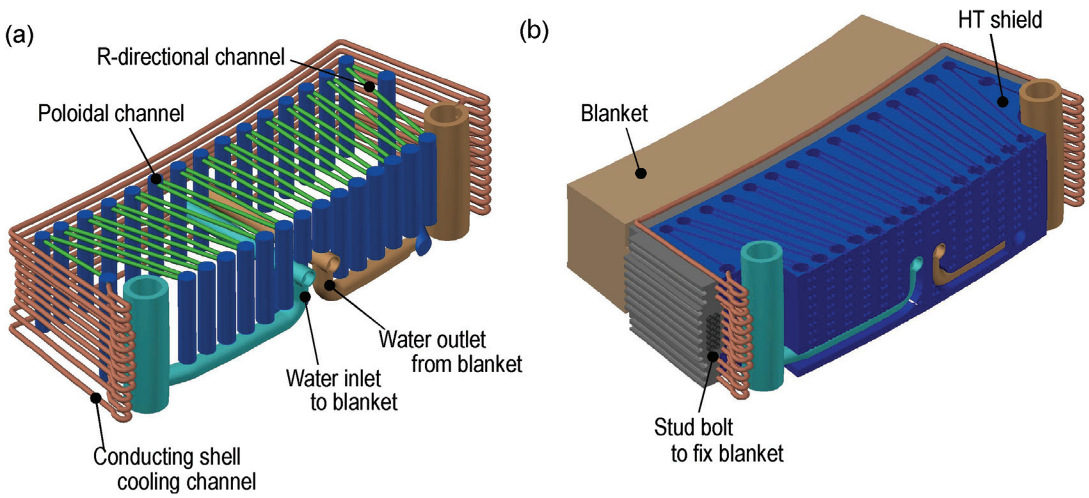

FIG.8 (a) Cooling channels of the shield (R-directional channels on a single layer is illustrated to improve visualization), and (b) cutaway view of the HT shield.

minimize the risk of irradiation damage of welding lines especially in the front part where neutron irradiation is more severe. Water outlet from the blanket is connected to the HT shield in series so as to reduce the complexity of the routing of coolant in the sector (FIG.8).

5. Low temperature (LT) shield

In order to remove the sector horizontally for maintenance, a support structure connecting the core components inside the HT shield and the port flange of the cryostat is needed. In SlimCS, the support structure acts as an additional shield by filling steel and water inside, which is called a low temperature shield (LT shield). The conceptual view of the LT shield is illustrated in FIG.9. Since the water filled in the LT shield has a temperature below 100ºC, the container of the shield is a nonpressure vessel. After the sector is removed from the cryostat as shown in FIG.1 and transported to the hot cell, the LT shield is separated from the assembly of the core components (core assembly) so as to ensure the accessibility for the replacement of blanket (FIG.9 (b)). A lot of piping for heat removal and tritium fuel recovery is arranged in the front of the LT shield, so that the LT shield is removable from the core assembly without cutting pipes installed in the sector.

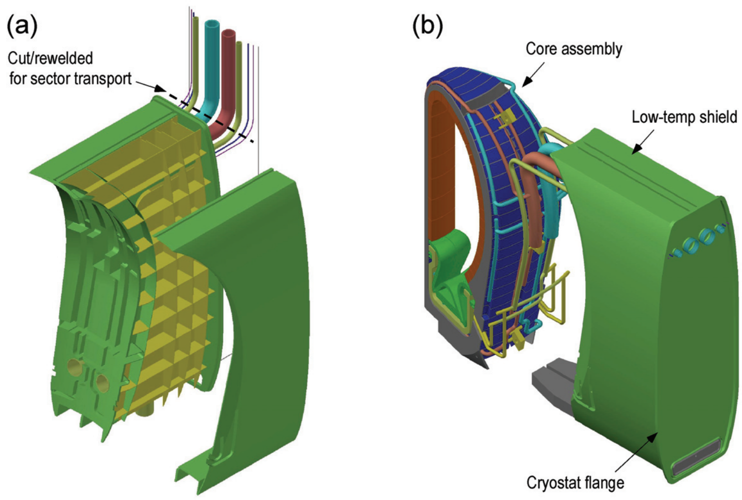

FIG.9 (a) Inner structure of the low temperature shield, and (b) arrangement of piping in front of the shield.

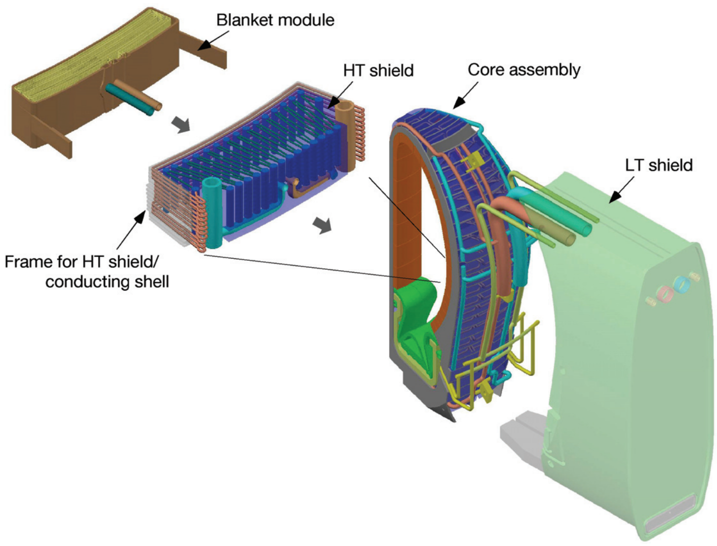

FIG.10 Conceptual view of the sector configuration

To summarize the core components presented in this paper, the configuration of the sector becomes as illustrated in FIG.10.

6. Summary

This study focused on the conceptual design of power core components of SlimCS, including blanket, conducting shell, HT shield, LT shield and the assembly of them. The result provides a positive prospect of a water-cooled DEMO reactor, although further careful design study is required to demonstrate the engineering feasibility.

On the other hand, we would like to stress the importance of these kinds of component designs toward DEMO, because some of them may be underestimated and has been left untouched in the fusion reactor design study despite the necessity of a technology jump from ITER. The HT shield treated in this paper is one such example. Water cooling is an issue that we need to pay more attention to, which is neither matured nor well understood. Instead, it is required to revisit the issues from the point of view of fusion power generation. The following are the coolant-issues that we have learned from the design study of SlimCS.

1) We chose the subcritical water condition (23 MPa, 290-360ºC) rather than the PWR water (15 MPa, 290-330ºC). However, the use of the subcritical water requires new material data supporting the material (F82H) compatibility. In addition, it creates new challenges related to plant engineering such as a steam generator. Therefore, generally speaking, the PWR water seems favorable for DEMO to utilize its matured engineering basis obtained in light water reactors. The problem is that the use of the PWR water may reduce TBR due to an increased cooling water retention in the blanket. Relation of water condition and TBR is an important design issue to be carefully studied.

2) Generally, TBR increases with decreasing neutron wall load. Since production of self-sufficient tritium is a requirement which is not easy to attain, each blanket module needs to be optimized to maximize TBR in accordance with neutron wall load which changes in the poloidal direction. At the same time, each blanket needs to be designed to output the cooling water with the same outlet temperature for reasonable heat engineering. The design methodology remains to be developed.

3) The water retention of the fusion DEMO plant amounts to about 1000 m^3, being three times as much as that of a light water reactor. The piping inside the cryostat and in the reactor hall and the HT shield are the main factors for such a high water retention. For the impact mitigation on a loss of coolant accident (LOCA), the cooling water system should be divided into several loops.

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