CONCEPTUAL DESIGN FOR THE POOL-TYPE SODIUM-COOLED FAST REACTOR

(3) Basic concept of reactor structure

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A conceptual study on a 600 MWe class pool-type sodium-cooled reactor was carried out for the purpose of establishing the basic concept of the plant prior to the conceptual design of the demonstration fast reactor. The report is a summary of the major design evaluations of the reactor structure.

1. OVERVIEW

The reactor structural concept of a 600 MWe class pool-type sodium-cooled reactor has been developed through design evaluations in terms of flow, seismic resistance, etc. In this study, the major design issues such as gas entrainment from the sodium coolant surface in the reactor vessel, coolant surface sloshing during an earthquake were examined and the prospect of feasibility for each was obtained.

2. REACTOR STRUCTURAL CONCEPT

In developing the basic concept of the plant, the reactor structure concept was established. FIG.1 shows the conceptual diagram of the reactor structure, FIG.2 shows the flow of conceptual study of the reactor structure, and TABLE 1 shows the main specifications of the reactor structure.

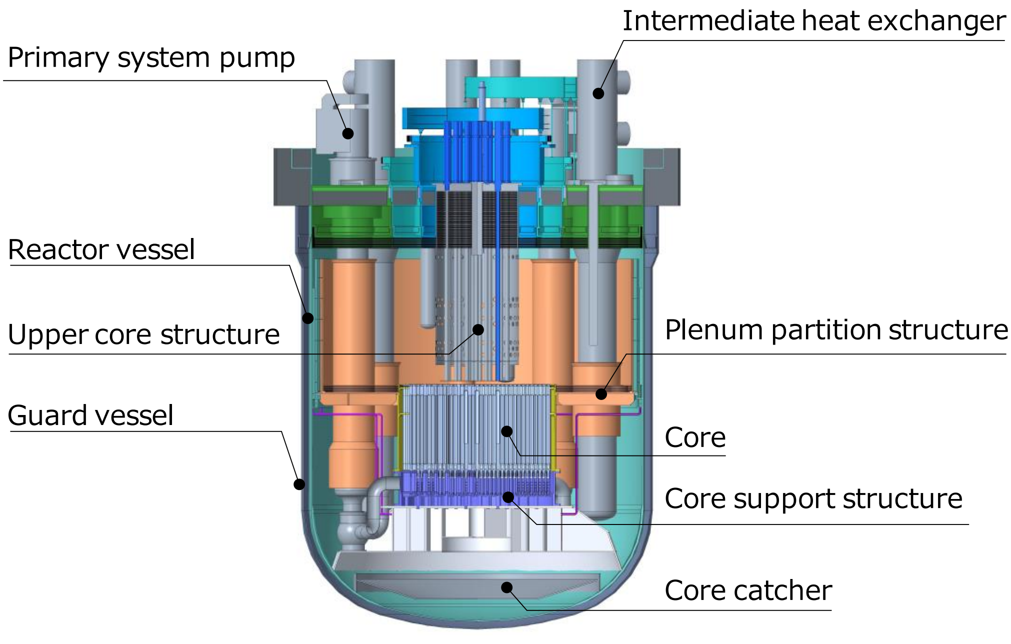

FIG. 1. Conceptual diagram of the reactor structure

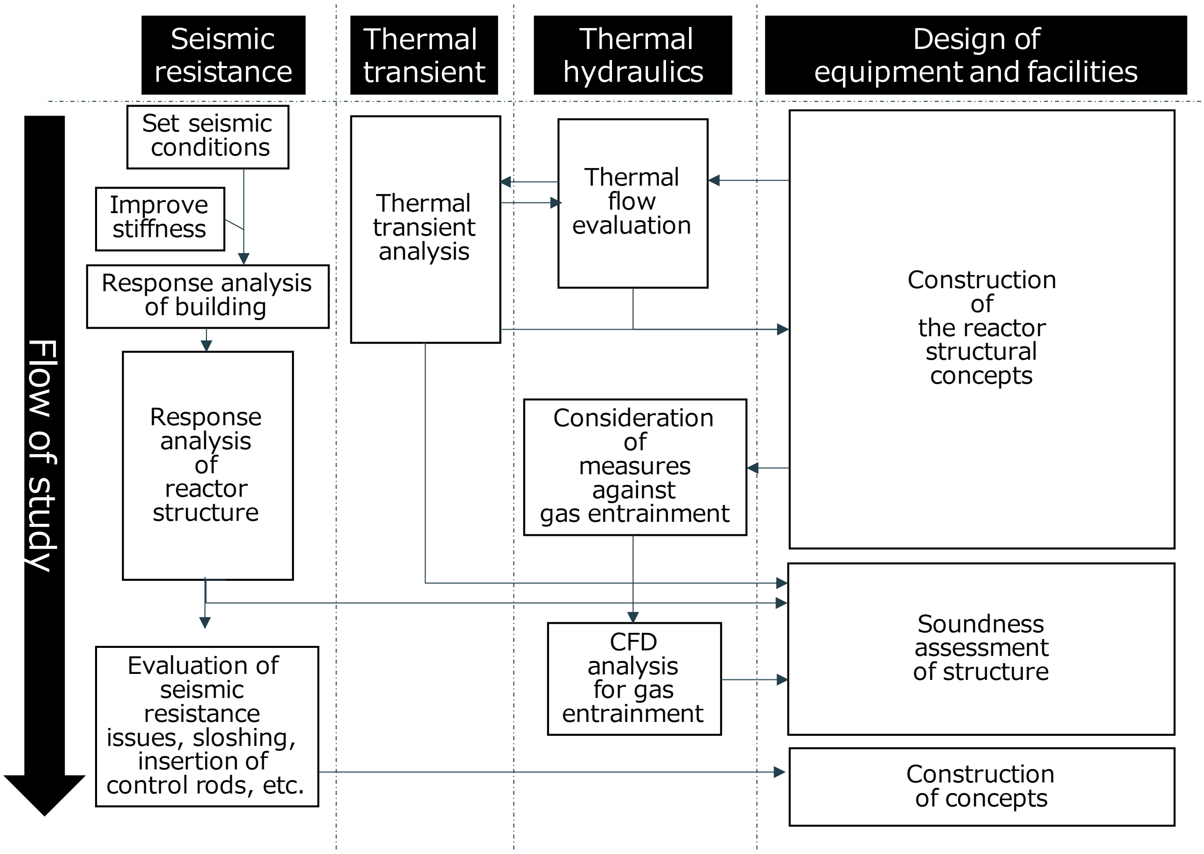

FIG. 2. Flow of conceptual study of the reactor structure

TABLE 1. MAIN SPECIFICATIONS OF THE REACTOR STRUCTURE

Reactor wall protection method	Low temperature sodium circulation
Main material	316FR steel
Operating temperature	High temperature section: 550°C Low temperature section: 400°C
Seismic condition	Seismic isolation building conditions
Number of intermediate heat exchanger	4 units
Number of pumps in primary system	3 pumps
Reactor vessel diameter	approx.16m
Axis length	apporx.20m

3. ADDRESSING KEY TECHNICAL ISSUES

The coolant system pressure of SFRs is lower than that of LWRs, and liquid sodium coolant free surface exists in the reactor vessel. In pool-type reactors, one of the major design issues related to thermal hydraulic aspects is the prevention of gas entrainment in the reactor vessel, which may pose a risk of heat transfer performance decrease and reactivity abnormalities, when the intermediate heat exchanger (IHX) placed inside the reactor vessel sucks gas from the sodium coolant free surface. FIG.3 shows the conceptual diagram of gas entrainment in the reactor vessel.

The evaluation was carried out using an evaluation tool called "StreamViewer" developed by Japan Atomic Energy Agency. Specifically, lines representing the center of vortex (vortex center line) extending from the inlet of the intermediate heat exchanger of a tank type SFR to the free liquid surface in the upper plenum were extracted, and the distribution of depressurization quantity along the vortex center lines were calculated. The gas core length was evaluated as the vortex center line length where the pressure was lower than the head pressure, and the occurrence of gas entrainment was judged. According to this evaluation, the vortex reaching the IHX inlet window was not generated, and gas entrainment was not expected.

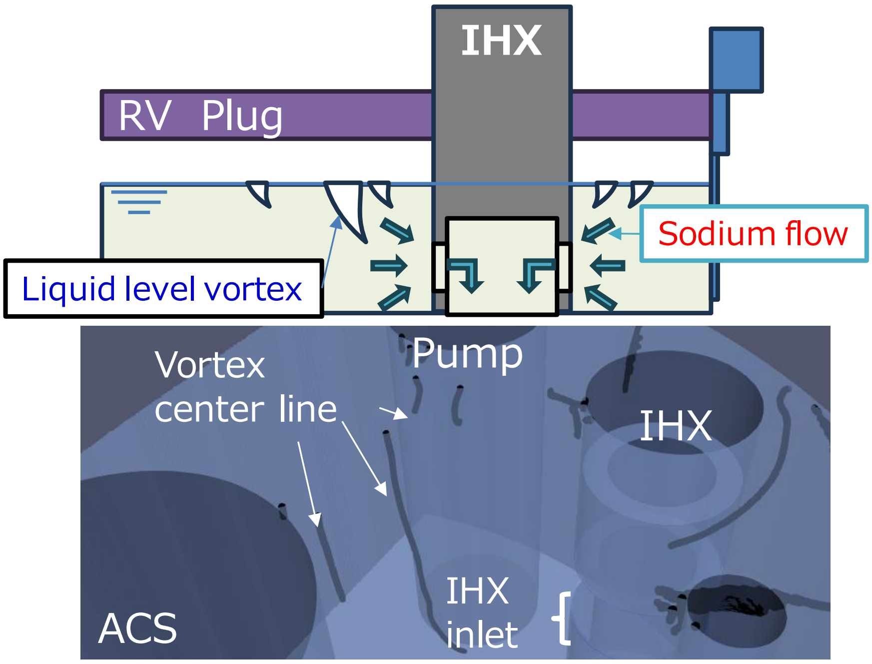

FIG. 3. Conceptual diagram of gas entrainment in the reactor vessel

Because the sodium coolant free surface exists in the reactor vessel, sloshing of the surface occurs during an earthquake. However, due to the adoption of the seismic isolation system, the seismic load is reduced, but sloshing of the surface tends to occur. In the conventional sloshing evaluation of SFR in Japan, the criterion is that the sodium coolant surface does not collide with the reactor vessel plug. However, due to severe seismic conditions even adoption of the seismic isolation system, it is necessary to estimate the impact load of the liquid surface of sodium on the reactor vessel plug.

The prospect that structural integrity would be feasible when the coolant surface collides with the upper part of the reactor structure was obtained through the numerical fluid analysis. Based on the evaluated impact load, the roof slab and thermal shielding layer, which are the impact parts, were designed to withstand the impact loads. As a result, soundness is expected to be ensured. FIG.4 shows the snapshot of impact load assessment by CFD, and FIG.5 shows the conceptual diagram of sloshing countermeasures.

Including these, design evaluations were conducted for complex technical issues such as structural integrity and thermal hydraulics, and the prospect of feasibility was confirmed.

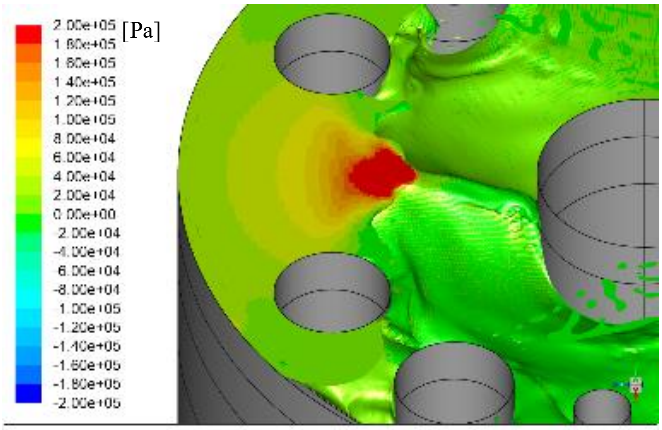

FIG. 4. Snapshot of impact load assessment by CFD

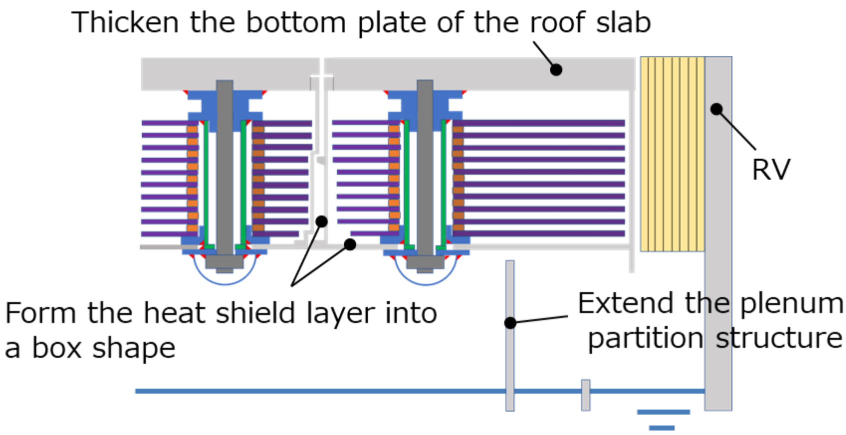

*FIG. **5**. Conceptual diagram of sloshing countermeasures*

4. CONCLUSION

Based on the reactor structural concept developed in this study, we are efficiently proceeding with conceptual design, which will begin in 2024.

ACKNOWLEDGEMENTS

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