Structural integrity assessment method utilizing 3D CFD analysis

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1. Introduction

As one of the core technologies for enhancing the safety of advanced small modular reactors (SMRs), an integrated analysis code combining fluid-structure and multidimensional-one-dimensional analysis has been developed. The three-dimensional thermal-fluid analysis of the SFR primary heat transport system, using chtMultiRegionFoam [1], focuses on the sodium pool, argon cover gas, nitrogen gas between the reactor vessel and containment vessel, head access area, and reactor vault cooling system. It also includes conjugate heat transfer within structural components such as reactor internal structures, upper internal structures, the vessel, and the reactor head.

To analyze the stress variations in the reactor vessel and containment vessel, solids4Foam was utilized [2-3]. The temperature changes in the structures are primarily influenced by the temperature of the sodium pool. The sodium pool temperature is affected by the core power conditions, as well as the heat removal rates through the intermediate heat transport system and the residual heat removal system. The core power distribution is applied based on the flow group of the fuel assemblies for each transient condition.

The heat transfer rate to the intermediate heat transport system or the residual heat removal system can be determined by inputting the heat removal rate into the intermediate heat exchanger and the decay heat exchanger when the heat transfer rate is known. For cases requiring coupled analysis, the GAMMA+ code [4] has been integrated. The reference reactor for establishing the coupled analysis framework is SALUS [5]. Detailed information on 3D-1D coupled analysis and fluid-structure interaction analysis can be found in reference [5].

2. Reactor Core Flow Analysis Modeling

2.1 Core thermal power modeling for SALUS

If there is assembly-specific thermal power information, it should be set in OpenFOAM's fvOptions in a table format according to time. Here, the assembly-specific power fractions are first calculated so that when the total core power information is given, the thermal power of each assembly can be determined. Based on the total core power, the fractions of each assembly are computed, and their thermal powers are derived from the time-dependent total core power and incorporated into fvOptions. A Python script, "coreFvOptions.py", was written to perform this task, using the assembly configuration information shown in Figure 1 as input.

In addition to handling thermal power of core assemblies, "coreFvOptions.py" script also sets the flow resistance for the porous media of the assemblies and nozzle orifices in fvOptions. Furthermore, a separate script, "flowGroup.py", was developed to define cellZone settings for flow groups within the core assemblies in the flow mesh.

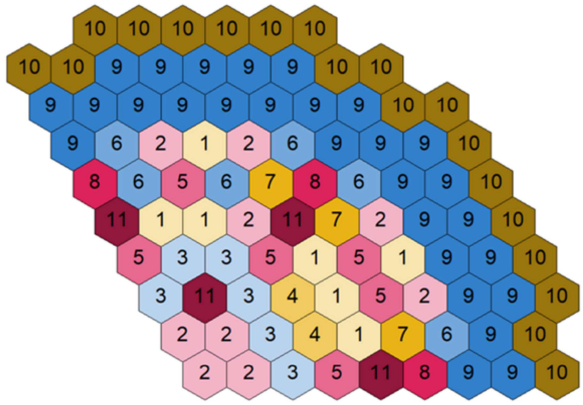

Fig. 1. Flow groups of core assemblies

2.2 Flow resistance modeling for core assemblies

The core assembly consists of fuel rod assemblies, reflectors, shielding assemblies, and control rod assemblies, and each assembly can be classified into flow groups as shown in Figure 1. Each assembly is composed of two porous medium regions: the lower part is the orifice, and the upper part is the assembly body. Flow and pressure loss data are provided according to the flow group, so the resistance of the flow paths in the orifice and body of each assembly can be determined. The flow resistance can be modeled using the Darcy-Forchheimer equation. To solve for the unknowns in the two terms, at least two different flow conditions and pressure loss data are required. However, only the flow and pressure loss data at the rated point are provided, making it difficult to determine the unknowns. The Darcy term is much smaller than the Forchheimer term, typically less than 10%, so the Forchheimer term is more significant and practically relevant in terms of resistance. The Darcy-Forchheimer equation is as follows, where the second term is the Forchheimer term.

$$\Delta P = \mu d L u + \frac{1}{2}\rho f L u^2$$

where, $\Delta P, \mu, d, L, u, \rho, f$ denote pressure loss, kinematic viscosity coefficient, Darcy coefficient, the flow direction length of the porous medium (the length of the cellZone in OpenFOAM when analyzing flow resistance in a porous medium), fluid velocity, fluid density, and the Forchheimer coefficient. In OpenFOAM, $f=2\Delta P/(\rho L u^2)$ must be given by user. The f value was calculated for the nozzle orifice and the assembly body regions.

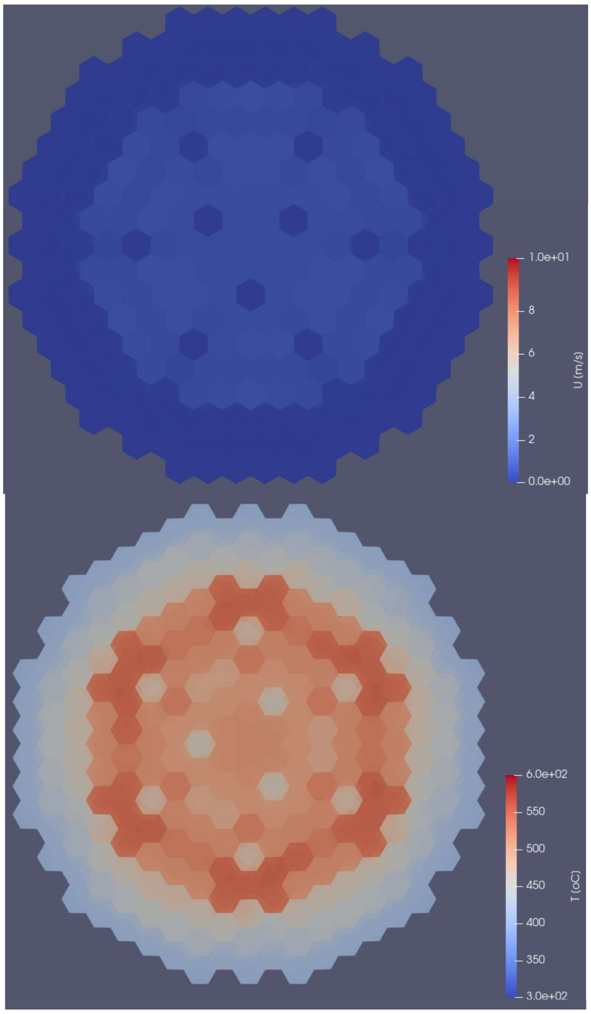

Fig. 2. Fluid velocity (top) and temperature (bottom) distribution at a core assembly section

Table I compares the flow rate design values with the values obtained from the analysis for each assembly. It can be confirmed that the modeling in this analysis is appropriate, as the deviations are small. The core is divided into cellZones based on the flow groups, and the Forchheimer coefficient is applied to each cellZone and inserted into the fvOptions file. Figure 2 shows the velocity and temperature distribution of the fluid within the assembly, where a uniform distribution can be observed.

Table I Comparison of mass flow rate design values and analysis values for flow groups

Flow Group	Difference(%)	Flow Group	Difference(%)
1	1.7	7	0.3
2	1.3	8	0.2
3	0.6	9(Reflector)	1.9
4	0.2	10(Shield)	1.8
5	0.1	11(CR)	1.9
6	0.3		

3. Analysis of Loss of Heat Sink in SALUS Unprotected Accidents

3.1 Accident analysis scenarios and conditions

The scenario selected for the accident analysis involves a situation where the reactor is not tripped despite the loss of heat sink, resulting in the continued asymmetry of the sodium pool. This corresponds to an Unprotected Loss of Heat Sink (ULOHS) accident, specifically a spurious one Intermediate Heat Transport System (IHTS) pump trip (ULOHS-1IP) [6]. In all target ULOHS accidents, the initial conditions represent the most conservative analysis results, with core output (102%), core inlet temperature (356°C), and core outlet temperature (514°C).

The transient analysis begins under normal conditions, and after 10 seconds, if one of the IHTS pumps stops, the heat removal rate through the IHTS decreases due to the reduced flow. The AHXs and FHXs of the DHRS, are fully activated 15.1 seconds and 20.1 seconds later, respectively, triggered by the core inlet temperature rise signal. The sequence of the accident is shown in Table II.

Table II Accident sequence of ULOHS-1IP

Sequence	GAMMA+	Description
Time of accident initiation (sec)	10.0	- Costdown start(Assumption of flow stoppage within 4 seconds)
Time of core inlet temperature rise signal occurrence (sec)	82.3	- AHX delay time 15.1s - FHX delay time 20.1s
AHX damper fully opening time (sec)	97.4	- AHX air damper fully opens after 15.1 seconds, including delay time
FHX blower full actuation time (sec)	102.4	- FHX blower operates at 100% after 20.1 seconds, including delay time

Figure 3 illustrates the core thermal power, including reactivity feedback, and is derived from the results of accident analysis using GAMMA+ [6]. Figure 4 compares the sodium temperatures of the hot pool and cold pool between the OpenFOAM and GAMMA+ analyses. In the OpenFOAM analysis, the temperatures at the core inlet and outlet are measured at single points, whereas the temperatures in the cold pool and hot pool represent volume-averaged values. After 100 seconds, the GAMMA+ analysis shows an increase in core inlet and outlet temperatures. However, in the OpenFOAM analysis, the temperature rise trend in the hot pool and cold pool is not clearly observed. The discrepancy between the system code and OpenFOAM analysis can be attributed to heat transfer through structures between the hot and cold pools, as well as heat transfer via the reactor vessel and reactor head. Even if the heat removal function through the intermediate heat transport system is lost, the temperature variation trend appears to be influenced by the magnitude of the core thermal power.

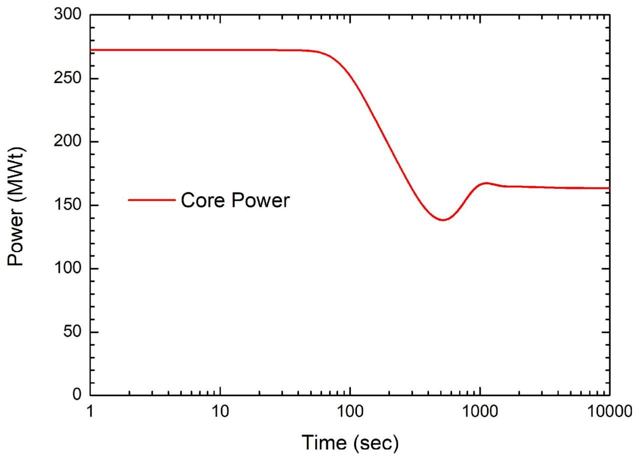

Fig. 3. Core power in case of ULOHS-1IP

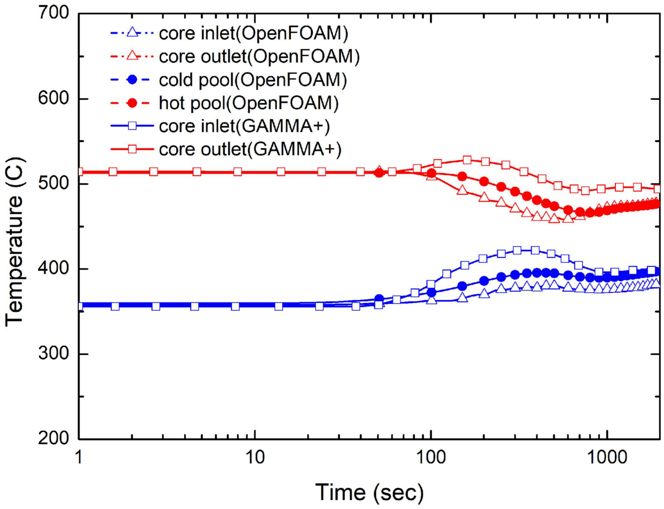

Fig. 4. Sodium temperature variations in case of ULOHS-1IP

Figure 5 shows the temperature distribution of the sodium pool over time, confirming that temperature differences occur in adjacent sodium regions depending on whether heat removal by the IHX is present.

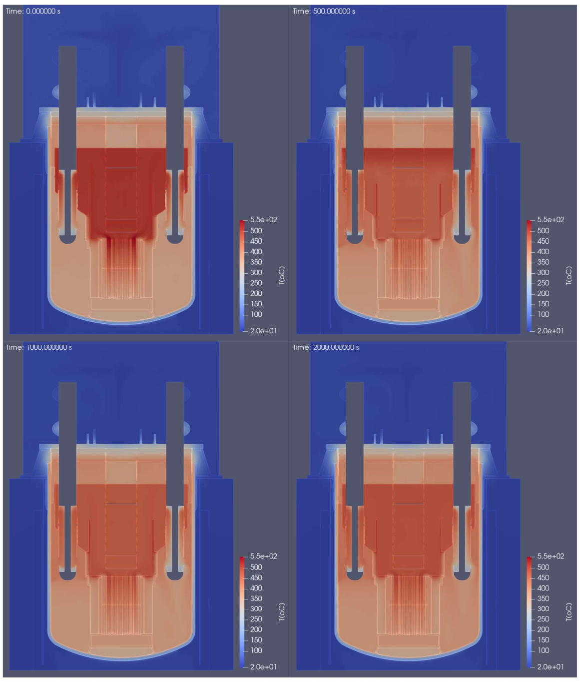

Fig. 5. Temperature fields of SALUS (from 0 s to 2000 s)

The temperature distribution on the inner surface of the reactor vessel adjacent to the IHX differs depending on whether heat removal is occurring. If heat removal is not functioning, a slight difference in the temperature gradient on the adjacent reactor vessel surface can be observed. The effect of this temperature gradient difference on the thermal stress of the structure can be confirmed through thermal stress analysis results.

Using the structural temperature data obtained from the conjugate heat transfer analysis, thermal stress was analyzed for the transient period from 0 to 2000 seconds. Figure 6 shows the distribution of thermal stress in the structure over time. Around 500 seconds, when the core thermal power is at its lowest, the thermal stress increases, but it subsequently decreases again. Figure 7 illustrates the thermal stress distribution on the inner surface of the reactor vessel. Depending on the presence or absence of heat removal, variations in the circumferential thermal stress distribution occur. It is observed that, compared to the IHTS system without heat removal, the wall thermal stress near the IHX in the IHTS system with heat removal is lower.

The results of the thermal stress analysis were provided as input to the mechanical structure team and used for the reactor vessel integrity assessment.

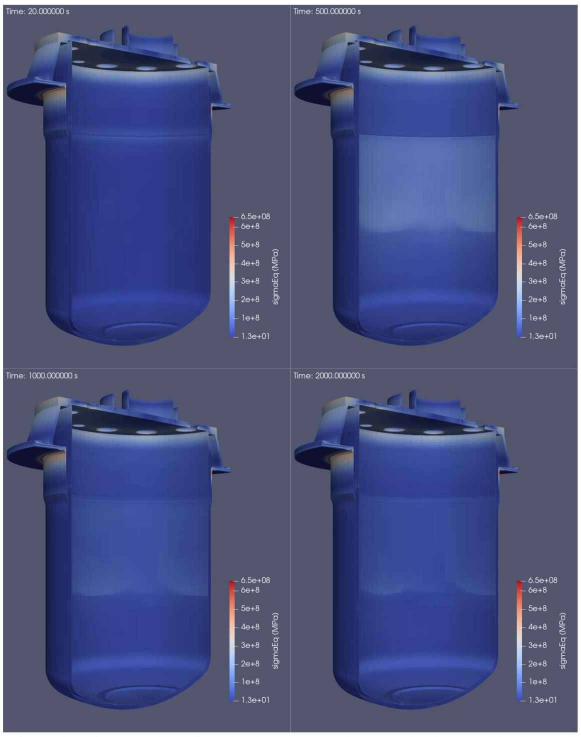

Fig. 6. Stress fields during transient of SALUS (Top:0 s(Left), 500 s(Right), Bottom:1000 s(Left), 2000 s(Right))

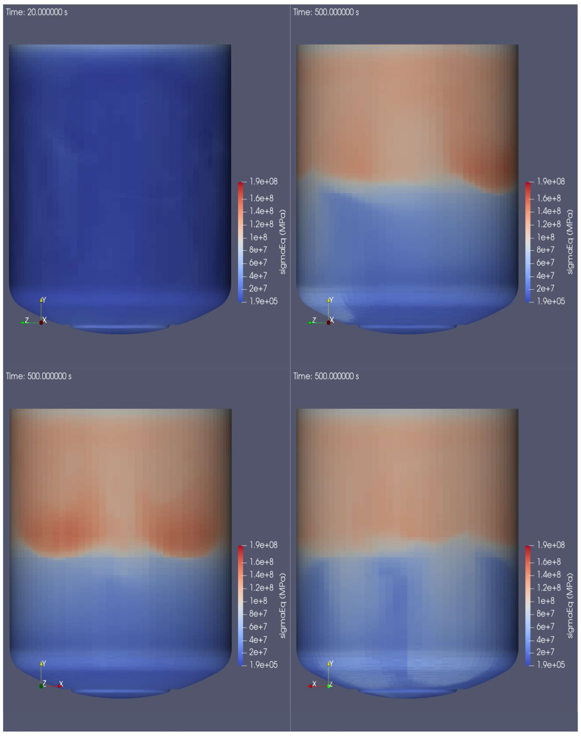

Fig. 7. Stress fields at RV inner surface during transient of SALUS (Top Left: 0 s, the others(different angle): 500 s)

The number of grids used in the 3D thermal-fluid analysis is approximately 15 million. When performing transient analysis for 2000 seconds, the computation was completed within 20 hours using a PC cluster.

3. Conclusion

A coupled analysis system was developed to analyze the thermal-fluid dynamics of the reactor. Using the temperature information of the structure obtained through conjugate heat transfer analysis, primary and secondary stress analyses were performed for the reactor vessel, containment vessel, and reactor head.

A script was written to easily set the core power and flow resistance for each assembly, and its appropriateness was verified by comparing it with design data. Using the developed analysis code, a transient analysis of the SALUS reactor was conducted for an unprotected accident scenario involving a single pump failure in the intermediate heat transport system. The characteristics of the thermal-fluid field and the trend of thermal stress changes in the structure were investigated. Through the developed analysis code, the structural integrity was conveniently assessed, confirming that design improvements could be identified in a short amount of time.

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