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SEISMIC ANALYSIS OF THE NUCLEAR REACTOR VESSEL CONSIDERING HYDRAULIC LOADS DURING OPERATING CONDITION

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ABSTRACT

A nuclear power generation is a large part of today's energy production. Many countries are promoting the construction of new nuclear power plants and are paying attention to nuclear power as a means of energy supply and demand. On the other hand, the issue of the risk of nuclear power generation certainly exists, and there are demands to abolish nuclear power plants. Therefore, in order to keep pace with this social movement, it is important to verify the structural integrity of earthquakes in the design of the nuclear reactor and internals. To verify the structural integrity for reactor system, seismic analysis is generally performed in the non-operating condition. However, the reactor internals are submerged in the coolant and the flow of the coolant makes hydraulic loads. In this research, seismic analysis was performed based on the dynamic characteristics extracted from scale-down model experiment. To reflect operating condition of reactor, periodic loads due to pump pulsation and random loads due to turbulent flow were calculated through hydrodynamic analysis. Finally, the safety of the seismic response considering hydraulic loads in the operating condition was confirmed by evaluating the margins of the reactor vessel and internals based on the American Society for Mechanical Engineers (ASME) code.

INTRODUCTION

Recently, as the intensity and incidence of earthquakes have increased, there has been enormous damages caused by earthquakes. The most recent example of an earthquake-related nuclear power plant accident is the Fukushima nuclear power plant accident that occurred in 2011. Nuclear accidents are classified as high-risk accidents because it can cause catastrophes for example, hydrogen explosion, leakage of radiation-polluted water, etc. In order to prevent such accidents, the earthquake-resistant design standard has been strengthened around the world.

To ensure the safety of nuclear reactor system, it is essential to analyse the nuclear vessel and internals correctly and diagnoses the risk. The nuclear reactor system consists of the reactor vessel and internals, pressurizer, steam generator, and reactor coolant pumps. The reactor vessel and internals include the reactor vessel (RV), core support barrel (CSB) assembly, upper guide structure (UGS) assembly, and Lower support structure (LSS). These components protect the fuel assemblies from external forces and maintain the nuclear functions.

However, the main natural frequencies such as the bending modes of reactor vessel and the internals are present at about 1Hz to 15Hz. Theses frequencies are included in the strong frequency band of the earthquake, which can adversely affect the response of the structure to earthquakes. Therefore, it is necessary to carefully analyse the responses of the structures to the seismic loads and accurately confirm the seismic margin.

Many researchers have studied the structural integrity of reactor through the finite element analysis (FEA) because it is difficult to conduct an experiment on a reactor directly, which is a huge structure. To obtain the seismic responses of the reactor, some researchers simulated the reactor vessel and internals as a beam model. The beam model can efficiently analyse the seismic responses of reactor by simplifying a structure having a complex shape and it is effective in seismic response that is dominantly influenced by the bending mode. However, the shell mode behaviour of the cylindrical reactor vessel cannot be simulated. In addition to the position of the stress acting on the structure can not be described in detail. Therefore, analysing a complex nuclear reactor system using a beam model has limited accuracy. Recently, to improve the accuracy of the analysis, the seismic analysis using a simplified 3D model, however, the accuracy of the stress response to seismic load is still limited due to sharp edges and non-uniformity of mesh of finite element model. Therefore, in order to obtain accurate stress responses, the finite element model should be constructed that reflects detailed shapes and accurate boundary conditions of reactor vessel and internals.

The reactor vessel contains coolant for cooling the fuel rods, and the reactor internals are submerged in the coolant. The structures interact with the coolant, which is the fluid-structure interaction such as added mass effect and gap fluid effect. Due to the added mass effect, the natural frequencies of reactor vessel and internals are decreased, and due to the gap fluid effect, the behaviours of structure are different by the coupling of structure and fluid. Therefore, the coolant inside the reactor also affects the seismic responses and must include the fluid in the seismic analysis.

The purpose of this research is to analyze the seismic response of reactor vessel and internals under the operating conditions. In order to apply the operating conditions, the hydraulic loads were considered. However, due to the huge analysis cost, it is almost impossible to perform time history seismic analysis considering hydraulic loads. Therefore, the responses for hydraulic loads were calculated as quasi-static analysis such as harmonic analysis and random vibration analysis. Deterministic hydraulic loads caused by the pump pulsation are periodic with the harmonic component of rotor rotation frequencies and blade-passing frequencies. So, the responses of deterministic hydraulic loads were calculated with harmonic analysis. Random hydraulic loads caused by the turbulent flow were converted to power spectrum density (PSD) to evaluate the responses of the structures in the view of energy. Finally, the seismic response and responses of hydraulic loads were added up and the seismic margin was evaluated based on the ASME code.

STRUCTURAL RESPONSE OF SEISMIC LOAD

In our previous research, 1/10 scale-down model for APR1400 was designed and produced. Then, both of the modal test in the air and the modal test in which the reactor was filled with water to take account of the fluid effect were conducted. So, the dynamic characteristics-natural frequencies and mode shapes-were intensively analyzed. The results of the modal test in air can be used to verify the geometries, material properties and boundary conditions of reactor vessel and internals and the results of the modal test which the reactor was filled with water can be used to demonstrate the fluid-structure interaction. Theses sequences of procedure of modal tests for obtaining the dynamic characteristics are shown in figure 1. Based on the results of the experiments, a finite element model for structural analysis was constructed and verified.

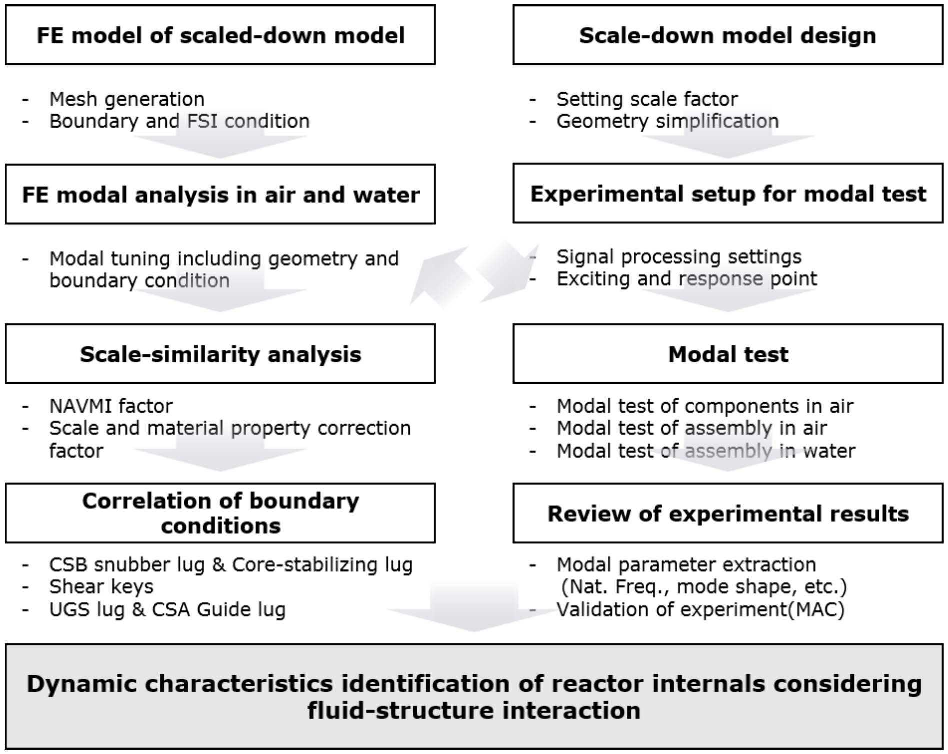


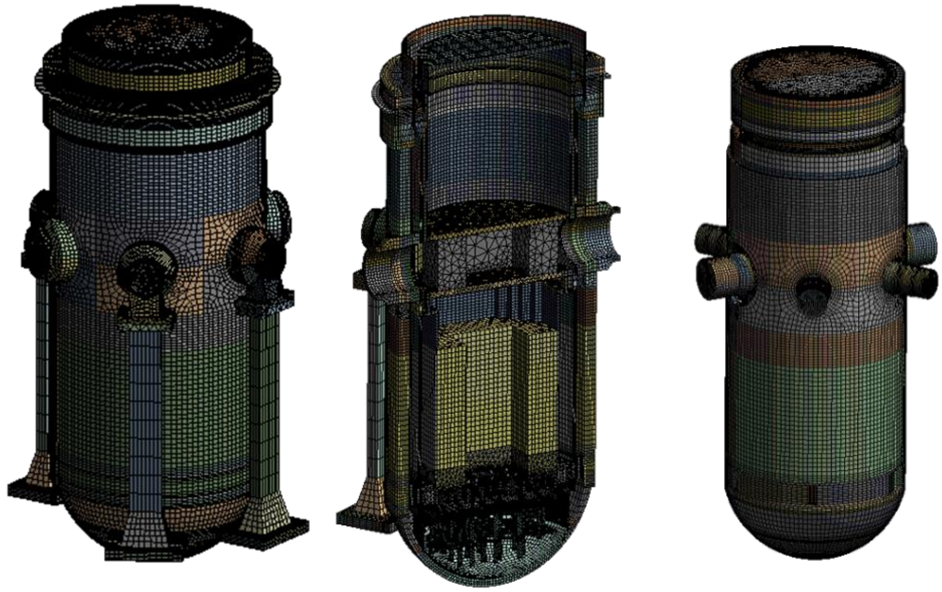
Figure 1. Procedure of modal tests for obtaining dynamic characteristics

Finite Element Model for Structural Analysis

As mentioned above, the finite element model was constructed based on the experimental results. However, since the results of the modal tests are the results for 1/10 scaled down model, additional work is required to predict the dynamic characteristics of the actual size reactor. So, by using the non-dimensional numbers, the correction coefficients such as size and material properties were calculated, and the similarity analysis was performed to predict the dynamic characteristics of the actual reactor. Finally, 3D finite element analysis model was constructed which shows the error within 10% comparing with the natural frequencies of the actual reactor calculated by applying the similarity theory. To construct the FE model, ANSYS Workbench v. 17.2 was used and the same commercial finite element analysis program was used for subsequent analysis. Figure 2 shows the constructed finite element model of the APR1400 reactor. Totally 2,290,389 elements were used in FE model: 602.702 solid elements (SOLID185 in ANSYS) and 1,687,687 fluid elements (FLUID30 in ANSYS). Further structural analyses such as seismic analysis and random vibration analysis were performed with this model.

Response of Seismic Load

In order to calculate the response of seismic load, it is essential to determine the seismic input signal. The American Society for Mechanical Engineers/American Nuclear Society Probabilistic Risk Assessment Standard 2009 (ASME/ANS PRA Std. 2009) recommends standard of the appropriate earthquake to perform a seismic analysis. If it is difficult to obtain a site-specific seismic input signal, it is recommended that NUREG/CR-0098 or similar seismic input signals be used. Therefore, the El Centro earthquake was selected because there is no site-specific seismic input signal for APR1400 reactor. Figure 3 shows the time history of the El Centro earthquake.



Total FE model (2,290,389 elements) | Solid elements (602,702 SOLID185 elements) | Fluid elements (1,687,687 FLUID30 elements)

Figure 2. Finite element model for structural analysis

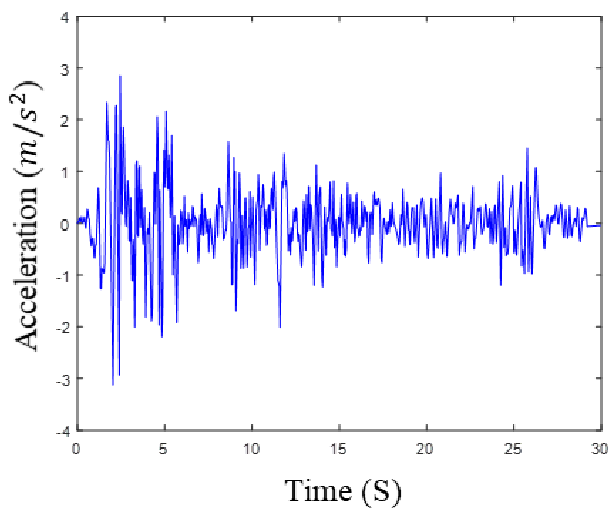


Figure 3. Time history acceleration of the El Centro earthquake

With the El Centro earthquake input signal, two type of seismic analyses were performed: The time history analysis (THA) and the response spectrum analysis (RSA). THA was performed with the El Centro earthquake signal of 30 seconds including the maximum acceleration of 0.3g, which is the seismic design criterion. RSA was performed spectral analysis using single point method. For the RSA calculation, the mode used for the analysis was used up to the 79th mode so that 90% of the effective mass was included. And the Square Root of Sum Square (SRSS) was used for the superposition method. Figure 4 shows the cumulative ratio of the effective mass for RSA. Figure 5 shows that both THA and RSA analysis show that the location of the maximum stress is same, and the maximum stress error is within 10%.

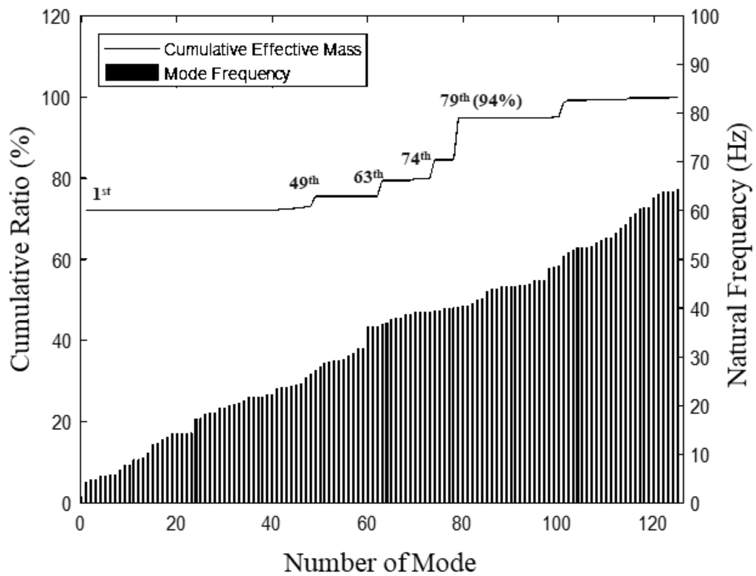


Figure 4. Cumulative ratio of the effective mass

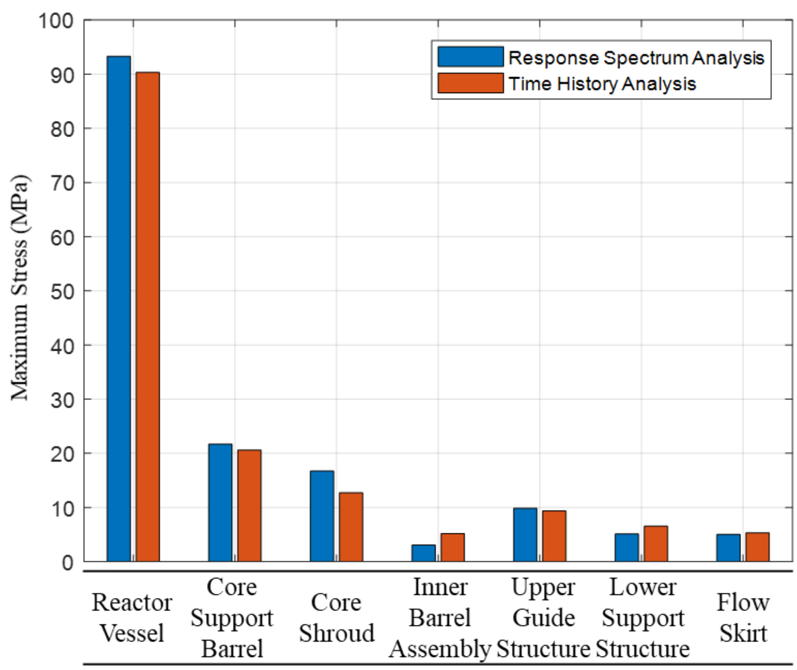


Figure 5. Maximum stress of THA and RSA analysis

STRUCTURAL RESPONSES OF HYDRAULIC LOADS

During operation, the reactor coolant makes hydraulic loads acting on the walls of reactor vessel and internals in the form of pressure. Theses loads can be divided into deterministic hydraulic load due to coolant pump pulsation and random hydraulic load due to turbulent flow generated by flow flowing between structures. Therefore, the responses by hydraulic loads were analysed in two ways.

Response of Deterministic Hydraulic Load

The deterministic hydraulic load is caused by the operation of the reactor coolant pumps. When the rotor of the pump rotates, the pump pulsation propagates in the form of an acoustic wave. And this propagated acoustic wave vibrates the reactor vessel and internals. The pump pulsation is determined by the rotor frequency (20Hz) and the blade-passing frequency (120Hz). The total deterministic hydraulic loads are calculated at six frequencies, consisting 20Hz and 120Hz and multiples: 20Hz, 40Hz, 120Hz, 240Hz, 360Hz, and 480Hz. The magnitudes of the pressure for each frequency components were obtained from the data measured by CVAP for the Palo Verde reactor, which is a forerunner of the APR1400 model. Total deterministic hydraulic loads were shown in figure 6. Theses pump pulsation pressures are applied to the inlet nozzles of reactor vessel. Because the deterministic hydraulic load consists of periodic components, harmonic analyses were performed according to the frequency components of the pump pulsation and the responses of each frequencies were superimposed based on displacement responses. Figure 7 and table 1 show the structural responses to the deterministic hydraulic loads. The responses at 360Hz and 480Hz, which are relatively high frequencies, are remarkable.

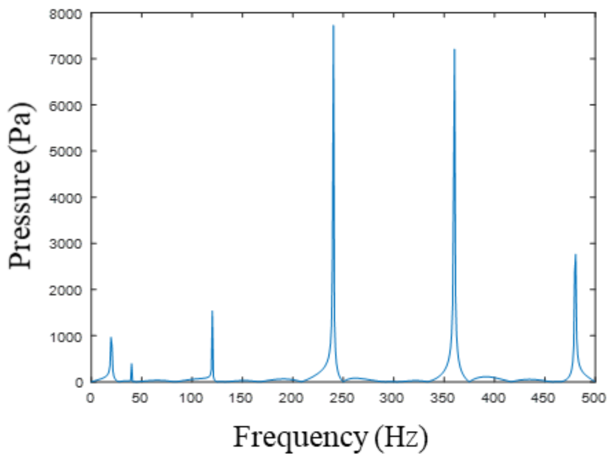


Figure 6. Pump pulsation pressures of deterministic hydraulic loads

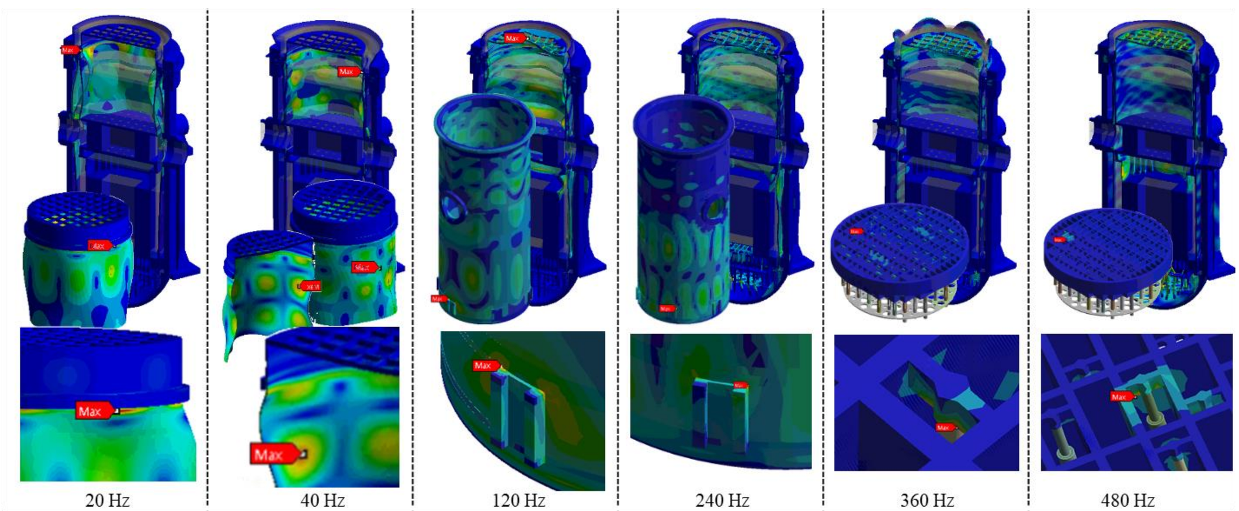


Figure 7. Responses of deterministic hydraulic loads

Table 1: Maximum stress responses of deterministic hydraulic loads

Responses of deterministic hydraulic loads (MPa)						
Case	20 Hz	40 Hz	120 Hz	240 Hz	360 Hz	480 Hz
Maximum Stress	2.08	0.69	0.09	2.54	1.90	1.25
	IBA	IBA	CSB	CSB	LSS	LSS

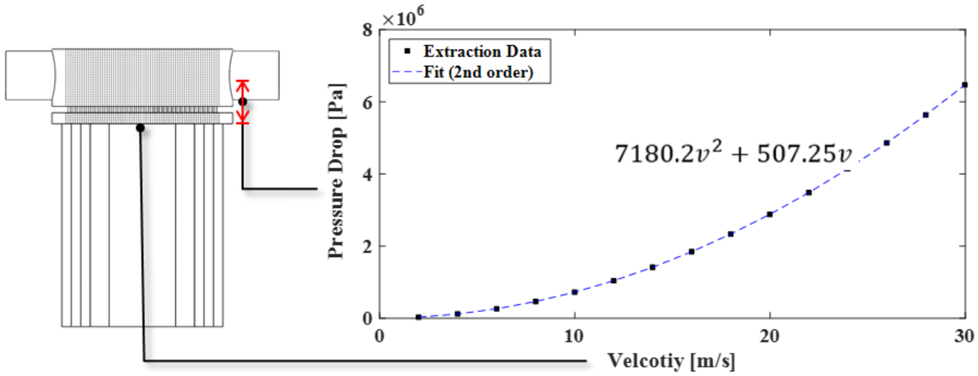


Figure 8. Fluid models for determining the porous coefficient of the UGS bundles

Finite Element Model for Fluid Dynamic Analysis

Random hydraulic load is caused by turbulent flow through the structures. In order to obtain the random hydraulic load, it is essential to perform a fluid dynamic analysis and calculate the pressure acting on the wall of reactor vessel and internals. To calculate the pressure acting on the wall precisely, structures such as lower support structure and flow skirt, which have a large influence on reactor internal flow, are modelled in detail. In addition, the mesh of the complicated flow path is refined to improve the accuracy of random hydraulic load. Finally, for the efficiency of the analysis, the pressure drop by UGS tube bundles is expressed as an equivalent porous model.

Equivalent porous model was constructed based on the Darcy's law as in equation 1 below:

$$\Delta \mathrm{p} = \frac{1}{2} C_2 \rho \Delta n v^2 - \frac{1}{\alpha} \Delta n \mu v \tag{1}$$

$\Delta \mathrm{p}$ is the pressure drop, C_2 is the loss coefficient, ρ is the density of fluid, Δn is the thickness of porous region, v is the velocity of inflow, and μ is the viscosity coefficient. Equivalent loss coefficient and permeability were determined by fitting the pressure changing the inflow velocity with a quadratic curve and comparing the coefficient of Darcy's law shown in figure 8. The fluid dynamic analysis model constructed through this series of processes is shown in figure 9.

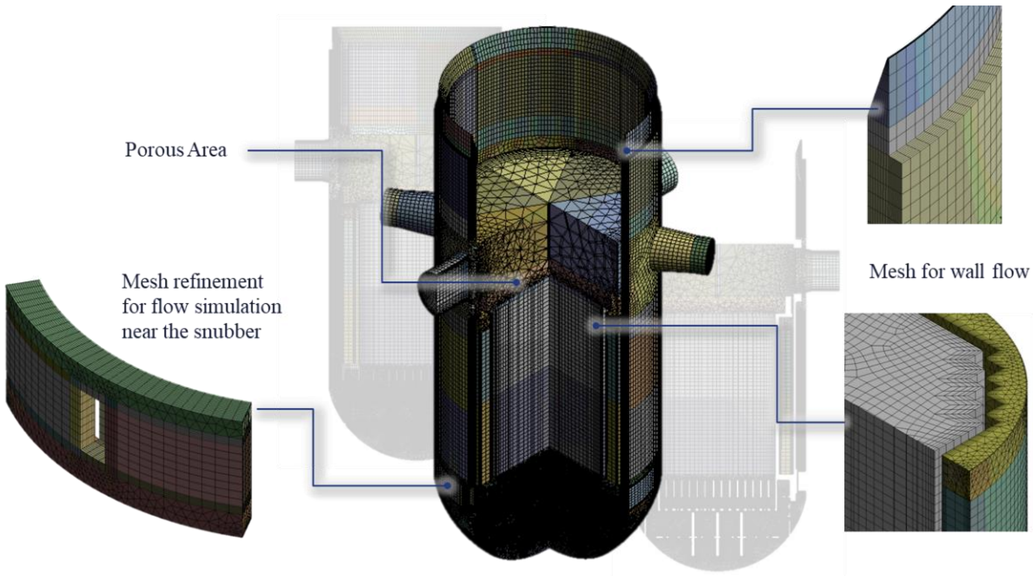


Figure 9. Fluid dynamic analysis model for random hydraulic load

Table 2: Detailed boundary conditions of fluid dynamic analysis

Temperature (°C)	Flow rate per pump (lb/s)				Outlet pressure (psi)	Turbulence intensity (%)	Turbulent viscosity ratio
	Inlet nozzle 1	Inlet nozzle 2	Inlet nozzle 3	Inlet nozzle 4			
290	11,500	11,500	11,500	11,500	2,250	5.0	10

The coolant fluid of the reactor assumed to be incompressible because the Mach number is less than 0.3 as the maximum velocity of the fluid inside the reactor is about 20m/s. The turbulent model was selected as the Reynolds-averaged Navier-Stokes equations (RANS) based Shear Stress Transport (SST) model which can represent not only the bulky flow but also the flow on the wall. The pressure rise of the reactor internal fluid by the reactor cooling pump was assumed to be equal to the total differential pressure in the flow system. Under normal 4-pump operating condition, the analysis was performed at the boundary conditions that the inlet flow rate and the outlet pressure is constant. The detailed fluid dynamic analysis boundary conditions are shown in table 2. The steady-state flow analysis was first performed to examine the appropriateness of the analysis results, and then the final time-history flow analysis was performed.

Response of Random Hydraulic Load

The pressures on the walls of the reactor vessel and internals were extracted by the fluid dynamic analysis and the graphs converted to PSD are shown in figure 10. Because the loads due to the turbulent flow are very irregular, the structural responses were derived from a random vibration analysis which obtains a probabilistic response to an irregular excitation force. For conservative evaluation, the stress response of random vibration 3-sigma probability was extracted. The maximum stress responses for each structure under random hydraulic loads are shown in table 3.

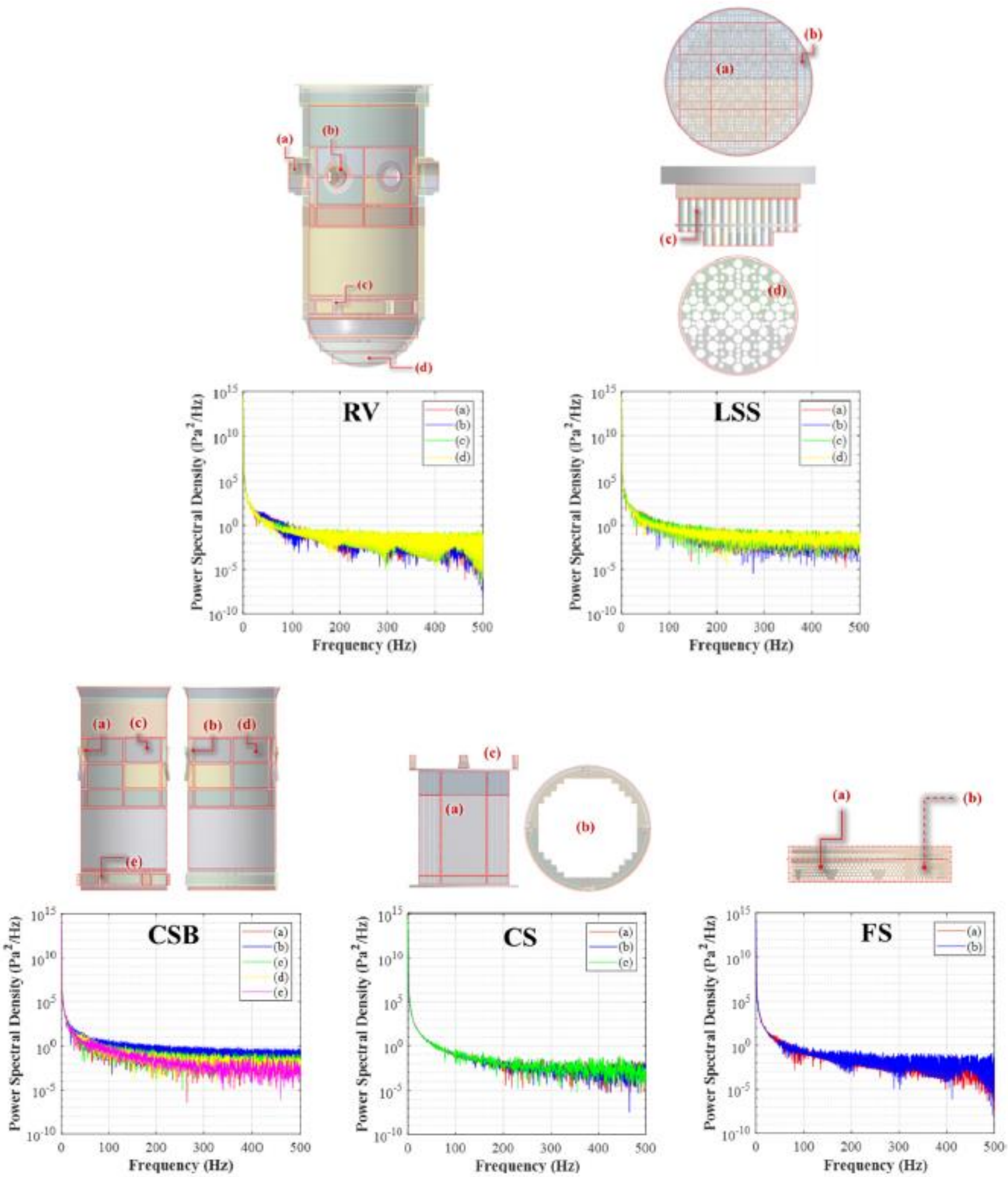


Figure 10. PSD data for each structure obtained from fluid dynamic analysis

Table 3: Maximum stress responses for each structure under the random hydraulic loads

Structure	Maximum stress (MPa)
Reactor Vessel (RV)	31.76
Core Support Barrel (CSB)	9.22
Lower Support Structure (LSS)	2.43
Core Shroud (CS)	10.36
Flow Skirt (FS)	0.32

Table 4: Final superimposed stress response and design margin of the APR1400 RVIs

Structure	Seismic analysis + Harmonic analysis + Random vibration analysis (MPa)	Design margin
Reactor Vessel (RV)	197.96	1.93
Core Support Barrel (CSB)	47.67	8.44
Lower Support Structure (LSS)	98.30	3.58
Core Shroud (CS)	83.83	4.37
Flow Skirt (FS)	14.53	29.97

EVALUATION OF MARGIN

All of the structural responses, such as response of seismic load, response of deterministic hydraulic load, and response of random hydraulic load were superimposed with two method. First, response of seismic load and response of deterministic load were superimposed based on the displacement. Displacement-based superposition method is able to superimpose stresses considering the vibration phase of a structure, so it is possible to reduce the conservative and derive a relatively accurate stress rather than simper stress superposition method. However, the random vibration analysis does not calculate the vector of the displacement, so it is impossible to use the displacement-based superposition method. Therefore, the final stress response of seismic load and hydraulic loads was derived through the simple superposition method. Then based on the ASME Code and the calculated final stress response considering seismic load and hydraulic loads, the design margin was calculated. As shown in table 4, the final superimposed stress response was expressed, and it was confirmed that APR1400 is sufficiently safe against the effect of an earthquake in the normal operating condition.

CONCLUSION

This research was conducted to verify the seismic safety of APR1400 RVIs in the normal operating condition. Structural response to pump pulsation and structural response due to turbulent flow were derived in consideration of the normal operating condition. And the seismic response was taken into consideration and finally seismic design margin was analyzed. Through this study, the safety of the APR1400 was confirmed by analysing the responses of the structures under various loading conditions.

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