Transactions, **SMiRT-24**
BEXCO, Busan, Korea - August 20-25, 2017
Division X

RESPONSE SPECTRUM ANALYSIS OF THE NUCLEAR REACTOR VESSEL AND INTENRALS VIA FINITE ELEMENT MODEL

Jong-beom Park[1], No-Cheol Park[2], Sang-Jeong Lee[1], Youngin Choi[3]

[1] Graduate Student Professor, Mechanical Engineering, Yonsei University, Korea
[2] Professor, Mechanical Engineering, Yonsei University, Korea
[3] Senior Researcher, Korea Institute of Nuclear Safety, Korea

ABSTRACT

Nuclear power generation system is one of the important energy generation systems. However, the system is very vulnerable to the external forces such as seismic waves. Especially, the nuclear reactor vessel and internals are very important because they protect the fuel assemblies directly. In order to design the structures more robustly, the seismic responses of the structures should be identified circumstantially. In this research, the seismic response analysis of an advanced pressurized water nuclear reactor 1400 (APR1400) was performed by using finite element (FE) model. It is very important to use the FE model that reflect the dynamic characteristics of the real system. To construct the valid FE model, the dynamic characteristics of reactor internal structures were identified by modal experiments. Then, the 3-D finite element model was designed to reflect the dynamic characteristics of the nuclear reactor vessel and internals. Finally, response spectrum analysis was performed and the seismic responses were obtained. From the results, the safety of the nuclear reactor vessel and internals were verified.

INTRODUCTION

Nuclear power generation system is a very widely used energy generation system. However, nuclear power plants are vulnerable to externals forces such as seismic loads because nuclear power system has a lot of internal structures having complex structures. Especially, the reactor vessel and internals are very important because they protect the fuel assembly directly. If an accident occurs, it can cause detrimental effects globally. Thus, the nuclear reactor vessel and internals have to be designed to endure the external forces.

To achieve that, structural integrity of the nuclear reactor system should be secured. Especially, in structural integrity assessment, seismic responses of the reactor internals should be determined. Prior to performing any seismic analysis of nuclear reactor internals, it is important to define their dynamic characteristics, such as the natural frequencies and the vibratory mode shapes, because they are used as bases for such seismic analyses. On that account, many researchers have studied the dynamic characteristics of reactor internals by considering the fluid–structure interaction (FSI) effect. Especially, Choi proposed a methodology for identifying the dynamic characteristics of the nuclear reactor system by adjusting analytical and experimental methods [1]. Similarly, in this research, the dynamic characteristics of a commercial nuclear reactor system was identified.

In common with the dynamic characteristics analyses, seismic responses were studied by many researchers. In order to consider the complex structures for the nuclear reactor internals, seismic analysis models have been developed based on lumped mass elements. Jhung focused on the detailed behaviors of commercial nuclear reactor internal structures [2]. Koo investigated the fluid effects on the dynamic behaviors of a liquid-metal fast-breeder reactor (LMFBR) and performed a seismic analysis of an entire nuclear power plant system [3]. However, most finite element models have not been supported by experimental verification. Also, the complex structures were simplified to reduce the analysis costs. In this research, to obtain the more exact seismic responses, 3-D FE model was used. Based on the model, response spectrum analysis was performed.

The main purpose of this research is that define the seismic response of reactor internals. The research was conducted based on the Korean commercial reactor – Advanced Power Reactor (APR) 1400. In this research, the seismic responses were determined based on FE analysis. To perform this seismic analysis, it is very important to construct the finite element model that reflect the dynamic characteristics of reactor internals exactly. Thus, modal experiments should be performed to extract the dynamic characteristics. However, they are hard to be performed in real system because modal experiments of real scale reactor internals needs a lot of costs and manpower. To solve that problems, scaled-down model was constructed and used to modal experiments. Based on the results of experiments, 3-D finite element model was constructed and the FE model was modified to real size by applying scale-similarity analysis method. Finally, using the FE model, response spectrum analyses were performed and the seismic responses were obtained. From the results, the safety of the nuclear reactor vessel and internals was verified.

Scaled-down model of the APR1400

In general, in order to construct a valid finite element model, it is essential to identify dynamic characteristics of a system exactly. Thus, finite element analysis should be performed along with modal experiments. However, in case of nuclear reactor internals, it is difficult to identify the dynamic characteristics of a real size model incorporating Fluid-Structure Interaction (FSI) since dynamic tests cannot be performed on the full size model in a laboratory. Accordingly, a scaled-down model was constructed for this study.

The scaled-down model should be small enough for the laboratory and reflect the dynamic characteristics of the real size nuclear reactor system. At the same time, it should be large enough to accommodate attached accelerometers and force transducers. Considering that conditions, 1/10 scaled-down model was constructed. Figure 1 shows the scaled-down model.

Figure 1. 1/10 scaled-down model of the APR1400

The model made of main components of reactor internals – reactor vessel, core support barrel assembly, upper guide structure assembly, inner barrel assembly. The major components were selected by preliminary FE analysis. The reactor internals have a lot of complex components, however, the selected components are mainly affect to the dynamic characteristics of the system. Thus, we designed the scaled-down model excluding structures that the natural frequency is beyond the results of seismic analysis. The scaled-down model is made of aluminum alloy 6061 for the sake of experimental control.

Scale-similarity analysis

Based on the results of the modal experiments, the finite element model was constructed. However, the model reflect the dynamic characteristics of the scaled-down model. Thus, the scale-similarity theory was used to identify the real scale model.

The added mass effect varies considerably according to the mode shapes and natural frequencies. Hence, it is necessary to add a non-dimensionalized added virtual mass incremental factor (NAVMI factor) to quantify the added mass effect with respect to the vibratory mode. The NAVMI factor (Γ) can be expressed by Equation (1), which is from previous research [4]–[8]:

$$\Gamma = \frac{1}{\zeta_f}\left[\left(\frac{f_O}{f_L}\right)^2 - 1\right] \tag{1}$$

where ζ_f, f_O, and f_L denote the density ratio of water to air, the natural frequencies of the structures in air, and the corresponding natural frequencies of the structures in water.

The material properties of the scaled-down model can be corrected using the elastic modulus correction factor (ε) and density correction factor (ζ_p). Equations (2)–(7) below are from previous research [9]:

$$\varepsilon = \frac{E_r}{E_s} \tag{2}$$

$$\zeta_p = \frac{\rho_r}{\rho_s} \tag{3}$$

where E_s, E_r, ρ_s, and ρ_r are the respective elastic moduli and densities of the materials used in the similarity model and the actual structure. The natural frequency (f_{Om}) of the similarity model with corrected material properties in air can then be written:

$$f_{Om} = \sqrt{\frac{\varepsilon}{\zeta_p}} f_{Os} \tag{4}$$

where f_{Os} denotes the natural frequency of the similarity model with uncorrected material properties in air. Also,

$$f_{Or} = \frac{1}{\kappa} f_{Om} \tag{5}$$

where f_{Or} denotes the natural frequency of the full-scale model with correct material properties in air, and κ is a geometric scale factor.

Assuming that the mode shapes in water are equivalent to the mode shapes in air, the NAVMI factor can then be expressed as follows:

$$\Gamma_s = \Gamma_r = \frac{1}{\zeta_s}\left[\left(\frac{f_{Os}}{f_{Ls}}\right)^2 - 1\right] = \frac{1}{\zeta_r}\left[\left(\frac{f_{Or}}{f_{Lr}}\right)^2 - 1\right]$$
$$with\ \zeta_s = \frac{\rho_{Ls}}{\rho_s},\ \zeta_r = \frac{\rho_{Lr}}{\rho_r} \tag{6}$$

where Γ_s, Γ_r, ρ_{Ls}, and ρ_{Lr} denote the respective NAVMI factors and fluid densities in the scale-similarity model and the full-scale model. Combining these equations, the natural frequencies of the full-scale model with different material properties can be obtained from the following formula:

$$f_{Lr} = \frac{f_{Os}}{\kappa}\sqrt{\frac{\varepsilon}{\zeta_p(1+\zeta_r\Gamma_s)}} = \frac{f_{Os}}{\kappa}\sqrt{\frac{\varepsilon\rho_s}{(\rho_r+\rho_{Lr}\Gamma_s)}} \tag{7}$$

As Eq. (7) indicates, the natural frequencies of the full-scale model with different material properties can be calculated from the scale factor, material property correction factors, and a NAVMI factor derived from FE analysis or a modal experiment.

Table 1: Scale-similarity analysis

Mode Shape (n, m)	Natural Frequencies [Hz]						
CSB	1/10 Scale-Similarity Model			Real Size Model			
					Water (Reactor Coolant)		
	Air	Water	NAVMI Factor	Air	From FEM	From NAVMI Factor	Discrepancy [%]
(1, 1) OOP	160.1	50	25.0	15.8	8.8	8.7	1.1
(1, 1) IP	160.1	67	19.4	15.8	11.1	10.7	3.6
(2, 1)	450.1	71	44.2	44.6	13.5	13.5	0.0
(3, 1)	457.0	82	81.2	45.2	15.5	16.2	-4.5
(4, 1)	744.8	225	26.9	73.7	38.9	39.4	-1.3

Boundary condition correction of the finite element model

The FE model is constructed based on the geometry of the scaled-down model. However, in the modal experiment process, the components are assembled and disassembled repeatedly to attach the accelerometer. Thus, some boundary conditions are hard to be considered in the scaled-down model. On that account, the specific boundary conditions have to be corrected to predict the dynamic characteristics of the real nuclear reactor system.

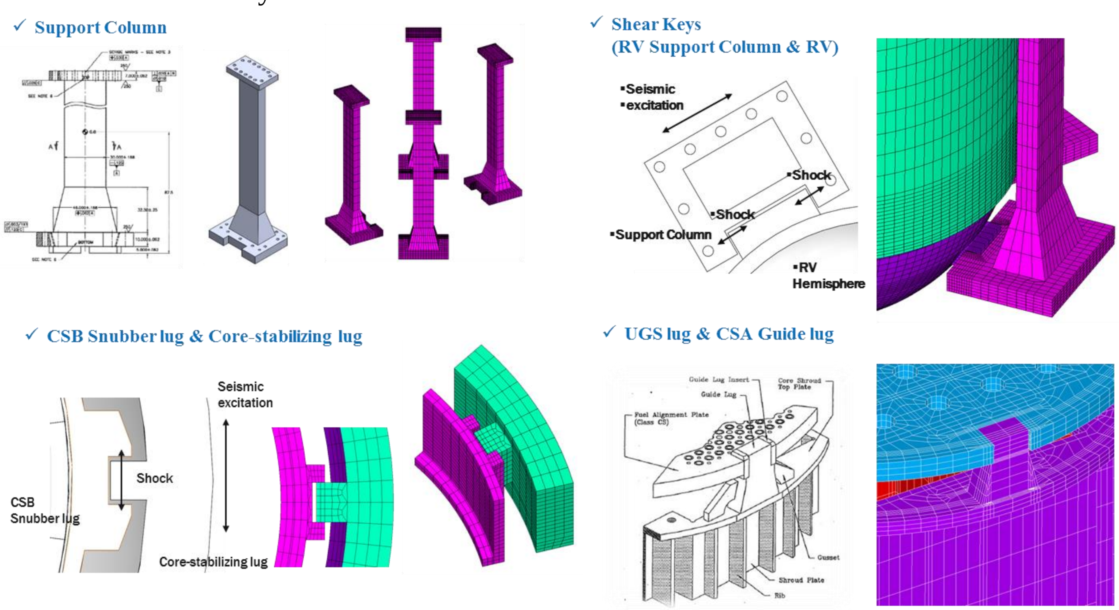

Figure 2. Boundary condition correction of the major components

Using the corrected FE model, the dynamic characteristics were extracted finally. Table 2 shows the results and figure 3 shows the FE model and the major mode shapes of the nuclear reactor internals.

Table 2: Scale-similarity analysis

Mode shapes	**Natural Frequencies (Boundary conditions Correction) [Hz]**
RV bending (x)	4.527
RV bending (y)	7.069
UGS bending with IBA OPP (y direction)	9.495
UGS bending with IBA OPP (x direction)	9.502
UGS bending with IBA IP (y direction)	14.67
UGS bending with IBA IP (x direction)	14.77
CSB shell mode (2, 1)	14.26
CSB shell mode (2, 1)	14.30

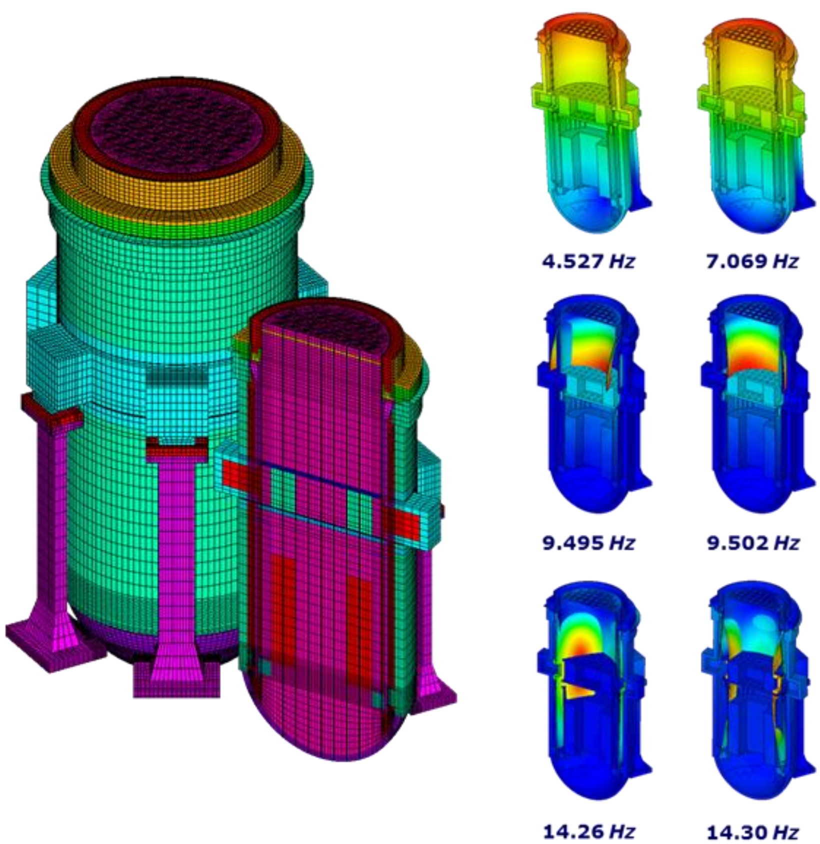

Figure 3. Finite element model and major mode shapes of the APR1400

Seismic analysis of the APR1400

In general, the seismic analysis is classified into response spectrum analysis(RSA) and time history analysis(THA). In this research, the response spectrum analysis was used to analyse the seismic response of the nuclear reactor internals.

In order to describe the real earthquake, the El Centro earthquake inputs were used. The acceleration inputs were inputted to the support columns. Assuming that the same input earthquake is applied to all supporting points, the spectrum analysis is performed using the single point method. There are various superposition methods in RSA. In this research, square root of sum square(SRSS) was used. Figure 4 shows the process of the RSA.

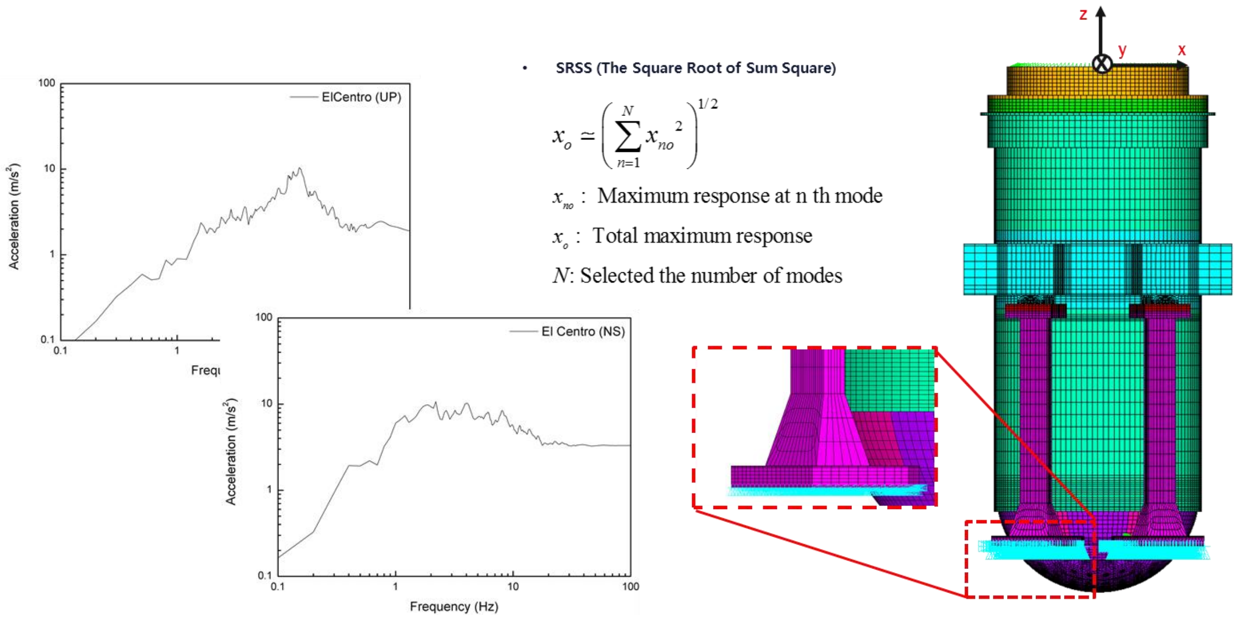

Figure 4. Process of response spectrum analysis

From the RSA, we extracted the stress values at the major components.

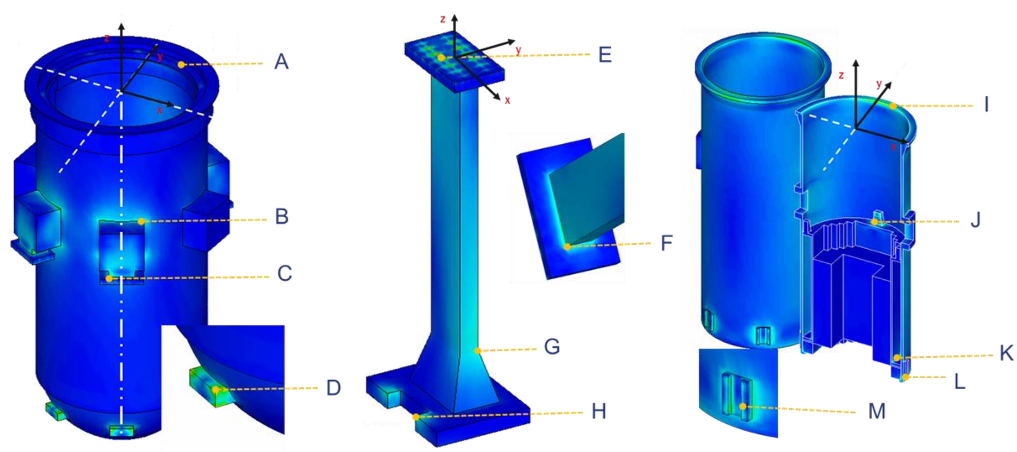

Figure 5. Stress contour Boundary condition correction of the major components

Figure 5 shows the stress contour of the major components. The bright regions have large stress values. Table 3 shows the stress values at the specific points.

Table 3: Stress values at the specific points of the major components

Position	Stress Values [MPa]			
	Sx	Sy	Sz	Equivalent
A	14.4	1.46	2.81	15.5
B	48.0	5.84	1.58	46.3
C	160.8	6.1	20.9	170.0
D	79.3	3.9	1.6	87.3
E	80.2	3.7	24.1	93.7
F	92.2	3.5	12.0	97.5
G	58.2	0.920	0.487	58.0
H	54.2	3.74	3.27	54.3
I	5.15	0.62	4.98	8.8
J	2.77	0.817	0.188	2.84
K	1.00	0.158	0.227	1.08
L	3.13	1.19	0.65	2.70
M	7.10	0.228	0.918	7.71

Generally, the tapered region and sharp corners shows the large stress. Otherwise, the core support barrel shows the lower stress compared to other structures because the core support barrel is clamped fully by the snubbers and the flange.

CONCLUSION

This research was progressed to analyze the seismic behavior of reactor internals. To achieve that, firstly, modal experiments were performed. Then, the finite element model was constructed to identify the dynamic characteristics of reactor internal structures. The validity of the model was verified by comparisons between experiments and finite element analysis. The constructed FE model was corrected based on the scale-similarity theory and the boundary conditions of the real system. Finally, by using the FE model, response spectrum analysis was performed and seiscmic responses of the major components were analyzed. This study can be developed to a reference in robust design of reactor internals.

REFERENCES

Choi Y.I., Lim S.H., Ko B.H., Park K.S., Park N.C., Park Y.P., Jeong K.H., and Park J.S., 2013. Dynamic characteristics identification of reactor internals in SMART considering fluid-structure interaction. Nucl. Eng. Des. 255, 202-211.

Jhung M.J., Hwang W.G., 1996. Seismic response of reactor vessel internals for Korean standard nuclear power plant. Nucl. Eng. Des. 165, 57-66.

Koo G.H. and Lee J.H., 2007. Development of Seismic Analysis Model and Time History Analysis for KALIMER-600. J. Earthquake Eng. Soc.Kor. 11 (3), 73-86.

Kwak M.K., 1991. Vibration of circular plates in contact with water. J. Appl. Mech. 58(2), pp. 480–483

Kwak M.K. and Kim, K.C., 1991. Axisymmetric vibration of circular plates in contact with fluid. J. Sound Vib. 146(3), pp. 381–389

Zhu, F., 1994. Rayleigh quotients for coupled free vibrations. J. Sound Vib. 171, pp. 641–649

Zhu, F., 1995. Rayleigh–Ritz method in coupled fluid–structure interesting systems and its applications. J. Sound Vib. 186, pp. 543–550

Amabili M., Frosali M and Kwak M.K., 1996. Free vibrations of annular plates coupled with fluids. J. Sound Vib. 191(5), pp. 825–846

S.H. Lim. 2011. Hydroelastic vibration analysis of reactor internals with fluid–structure interaction through scale-similarity model. Yonsei Univ.