Numerical analysis of reactor internals under hydrodynamic loads

Da Hye Kim [a], Yoon-Suk Chang [a*] and Myung Jo Jhung [b]
[a] *Department of Nuclear Engineering, Kyung Hee University, Yongin, Korea*
[b] *Korea Institute of Nuclear Safety, Daejeon, Korea*
[*] *Corresponding author: yschang@khu.ac.kr*

1. Introduction

Reactor internals installed in a pressure vessel have been exposed to harsh environment such as high neutron irradiation and temperature with complex fluid flow. As the increase of operational years of NPPs(Nuclear Power Plants), possibility of functional loss of the reactor internals is increased due to degradation caused by radiation embrittlement, thermal aging, fatigue, corrosion and FIV(Flow-Induced Vibration) etc. In practice, defects were detected at core support structure as well as upper and lower parts of structural assembly in European and United States NPPs.

Recently, in a GALL(Generic Aging Lessons Learned) report[1], US NRC(Nuclear Regulatory Commission) identified reactor internals as a high priority component and addressed relevant management programs. In Korea, similar activities have been conducted for long-term operation beyond design lifetime but most of them were limited to qualitative evaluation based on examination and maintenance programs. Therefore, not only to reduce repair and replacement efforts but also to secure the stability of NPPs, necessity for development of quantitative evaluation technique as well as establishment of preventive action plan and management procedures is on the rise[2].

The FIV[3] represents the structural vibration phenomenon induced by liquid flow and generally occurs at contact surfaces. In the present paper, our concern is focused on hydrodynamic loads anticipated to contribute the FIV among diverse degradation mechanisms. Characteristics of the reactor internals are examined and CFD(Computational Fluid Dynamics) analyses are carried out to investigate their structural behaviors. Subsequently, FE(Finite Element) analyses are performed to assess structural integrity of the reactor internals.

2. Analysis Methods and Results

2.1 Subcomponent Identification

The reactor internals usually designate all the equipment and parts inside of a reactor pressure vessel except for fuel assembly, control rod assembly, in-core instrument and surveillance capsule assembly.

In this context, we selected a typical pressurized water reactor of which schematic is depicted in Fig. 1. Based on inherent functionalities, the reactor internals were classified into 6 kinds of subcomponents; CSB(Core Support Barrel), UGS(Upper Guide Structure), CSA(Core Shroud Assembly), LSS(Lower Support Structure), GSSS(Guide Structure Support System) and CEA(Control Element Assembly) shroud.

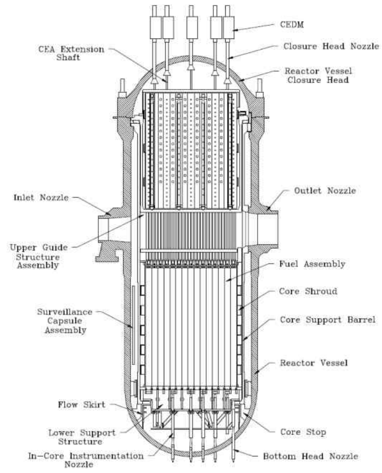

Fig. 1 Schematic of typical PWR internals adopted in this research[4]

2.2 CVAP Test Condition

The CVAP(Comprehensive Vibration Assessment Program) is a well-known vibration evaluation plan fulfilled during virtual operation and test operation by considering the FIV at normal and excessive situations. By applying the CVAP, it is able to check the integrity and stability margins of the reactor internals, and to use them as design input data of the same type NPPs. In this research, for the numerical analyses, a representative CVAP test condition summarized in Table 1 was selected.

Table 1 A representative CVAP test condition

Condition	Temperature (^{o}C)	RCP operation			
		1A	1B	2A	2B
Steady State	295.5	O	O	O	O

2.3 Numerical Analysis

A detailed 3-dimensional CAD model was developed based on design documents of the typical NPP[4~7]. Prior to main analyses, the geometry was optimized by considering numerical analysis capability; one is modeling of important subcomponents and the other is simplification of the model.

Fig. 2 represents optimized numerical analysis model which consists of 1,249,818 nodes and 6,398,678 elements with the mesh quality of 98%.

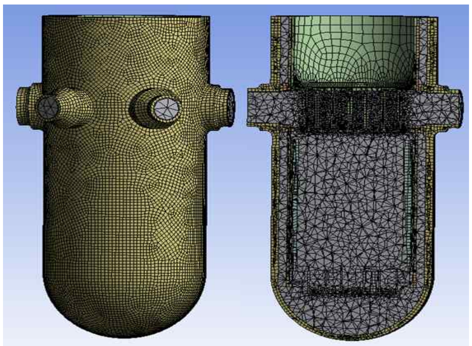

Fig. 2 Numerical analysis model

Two types of numerical analyses were carried out by using a commercial program[8].

At first, in case of periodic hydrodynamic analysis, very low stress values of 1MPa, approximately, were calculated as depicted in Fig.3.

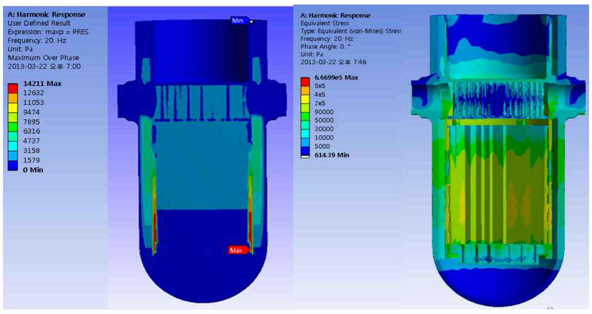

Fig. 3 Acoustic pressure and von Mises stress distribution

Secondly, in case of random hydrodynamic analysis, Shear Stress Transport(SST) option that is appropriate for the steady state condition was employed. As shown in Figs. 4 and 5, UGS and CSB have relatively high stress distribution. The order of high stress levels were UGS > CSB > CSA > LSS.

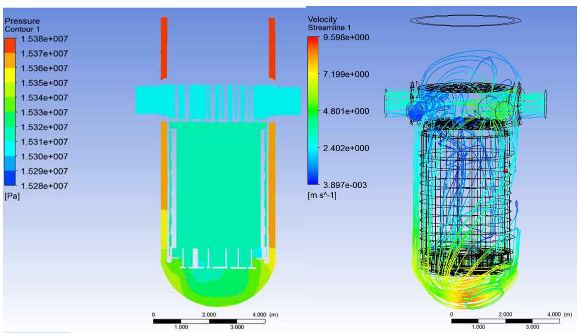

Fig. 4 Pressure and velocity distribution under random hydrodynamic load

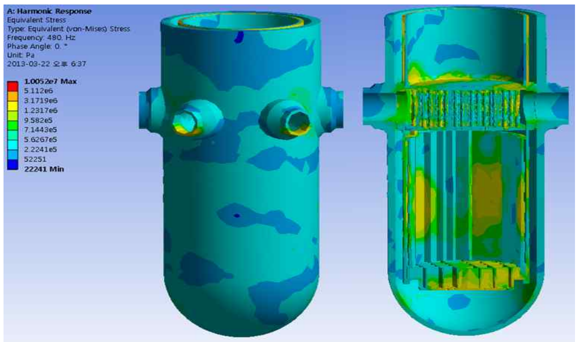

Fig. 5 von Mises stress distribution under random hydrodynamic load

3. Conclusions

In the present study, six kinds of major equipments of a typical reactor internals were identified by incorporating recent research trend. Based on this, detailed numerical models were developed and used for establishment of optimum analysis methodology subjected to hydrodynamic loads. As a result, stress values of the major equipments were calculated through the acoustic-structure analysis under periodic hydrodynamic load and the turbulence-structure analysis under random hydrodynamic load. The numerical analysis scheme can be used for development of preventive action plan and management procedures of the reactor internals.

REFERENCES

[1] USNRC, 2010, “Generic Aging Lessons Learned (GALL) Report,” NUREG-1801, Rev.2.

[2] Jhung, M.J., Yu, S.O., Lim, Y.T., 2011, "Dynamic Characteristics of a Partially Fluid-filled Cylindrical Shell," Nuclear Engineering and Technology, Vol.43, No.2, pp.167-174.

[3]Blevins, R.D., 1977, “Flow-Induced Vibration,” Van Nostrand Reinhold.

[4] Internal communication with KINS. 2012.

[5] Kim, Y. S., Kim, K. H. and Lee, J. H., 2010, “Hydraulic Analysis Methodology of Reactor Vessel Internals for Comprehensive Vibration Assessment Program,” Transactions of the Korean Nuclear Society Autumn Meeting, pp. 449~450 (in Korean).

[6] Ko, D. Y., Kim, K. H. and Kim, S. H., 2011, “Selection Criteria of Measurement Locations for Advanced Power Reactor 1400 Reactor Vessel Internals Comprehensive Vibration Assessment Program,” Transactions of the Korean Society for Noise and Vibration Engineering, Vol. 21, No. 8, pp. 708~713 (in Korean).

[7] Autodesk inventor professional 2011.

[8] ANSYS Workbench ver.14.0., 2012.