Structural Engineering and Mechanics, *Vol. 51, No. 6 (2014) 1005-1016*
DOI: http://dx.doi.org/10.12989/sem.2014.51.6.1005

Numerical study on fluid flow by hydrodynamic loads in reactor internals

Da-Hye Kim[1a], Yoon-Suk Chang[*1] and Myung-Jo Jhung[2b]

[1]*Kyung Hee University, Seocheon-dong, Giheung-gu, Yongin 446-701, Korea*
[2]*Korea Institute of Nuclear Safety, Guseong-dong, Yuseong-gu, Daejeon 305-338, Korea*

(Received December 17, 2013, Revised April 28, 2014, Accepted June 19, 2014)

Abstract. Roles of reactor internals are to support nuclear fuel, provide insertion and withdrawal channels of nuclear fuel control rods, and carry out core cooling. In case of functional loss of the reactor internals, it may lead to severe accidents caused by damage of nuclear fuel assembly and deterioration of reactor vessel due to attack of fallen out parts. The present study is to examine fluid flows in reactor internals subjected to hydrodynamic loads. In this context, an integrated model was developed and applied to two kinds of numerical analyses; one is to analyze periodic loading effect caused by pump pulsation and the other is to analyze random loading effect employing different turbulent models. Acoustic pressure distributions and flow velocity as well as pressure and temperature fields were calculated and compared to establish appropriate analysis techniques.

Keywords: hydrodynamic loads; pump pulsation frequency; reactor internals; turbulent models

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 irradiation embrittlement, thermal aging, fatigue, corrosion and flow-induced vibration etc. In practice, several failures of reactor internals have been reported world widely; lots of defects were detected at core support structure as well as upper and lower parts of structural assembly in European and United States NPPs. Also, control rod guide thimble support fin and BFB (Baffle Former Bolt) were damaged and replaced in a Korean NPP.

Recently, in a generic aging lessons learned report (2010), US NRC (Nuclear Regulatory Commission) identified reactor internals as one of high priority components 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

*Corresponding author, Professor, E-mail: yschang@khu.ac.kr
[a]Ph.D. Student, E-mail: dhikm@khu.ac.kr
[b]Ph.D., E-mail: mjj@kins.re.kr

http://www.techno-press.org/?journal=sem&subpage=8 ISSN: 1225-4568 (Print), 1598-6217 (Online)

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 (Jhung *et al.* 2011).

Thermal hydraulic assessment of the reactor internals is prerequisite to predict performance of intended functions which rely on safety features of design as well as operation of NPPs. For instance, a scale-down SMART (System-integrated Modular Advanced ReacTor) model was used for experimental investigation of flow-induced pressure variation under normal operating conditions (Lee *et al*. 2012). Also, structural response analyses of core support barrel and lower support structure of APR1400 reactor were carried out to demonstrate margins compared to the design acceptance criteria (Ko and Kim 2013) and CFD (Computational Fluid Dynamics) analyses were conducted to show proper design change of AP1000 reactor vessel upper internals (Xu *et al*. 2013).

Among CFD analyses implemented up to recently, particularly, effects of periodic loading due to pump pulsation and random loading due to fluid flow have been examined; with regard to the former, while few researches have been published, pressure variation induced by different pump operating conditions could be predicted (Kim *et al*. 2012). With regard to the latter, diverse results were reported such as flow patterns of a PWR reactor vessel upper plenum (Wu *et al*. 2012), flow distribution at the core inlet of a system-integrated modular advanced reactor (Bae *et al*. 2013), and thermal-fluid behavior in core region of a high temperature engineering test reactor under steady state and transient conditions (Tsuji *et al*. 2014).

In the present paper, inspired by the aforementioned researches along with further experimental and numerical activities (Gu *et al.* 2010, Hermansky and Krajcovic 2011, Kao *et al.* 2011, Kim *et al.* 2013, Ko *et al.* 2011, Park *et al.* 2012), a series of numerical study is performed for reactor internals of a representative Korean pressurized water reactor to investigate adequacy of long-term operation. An integrated model of the reactor internals is developed together with a VR (Virtual Reality) prototype and used to carry out numerical analyses subjected to hydrodynamic loads. Both the periodic loading effect caused by pump pulsation and random loading effect employing different turbulent models are analyzed. Acoustic pressure distributions and flow velocity as well as pressure and temperature fields are calculated and compared for establishment of appropriate techniques to decide critical locations in the future.

2. Characteristics of reactor internals

2.1 Design features

The reactor internals designate in general all the equipment and part inside of a reactor pressure vessel (RPV) except for fuel assembly, control rod assembly, in-core instrument and surveillance capsule assembly. These can be classified into the core support barrel (CSB) assembly and upper guide structure (UGS) assembly, which are installed and withdrawn as each assembly. The CSB assembly includes the CSB, the lower support structure (LSS) and in-core instrumentation nozzle assembly and the core shroud assembly (CSA). The CSB is a right circular cylinder supported by a ring flange from a ledge on the reactor vessel. It carries the entire weight of the core. There are six snubbers, which are equally spaced, at the bottom of the CSB to prevent the torsional motion of the internals. The LSS transmits the weight of the core to the CSB by means of a grid beam structure. The core shroud surrounds the core and minimizes the amount of bypass flow. The UGS

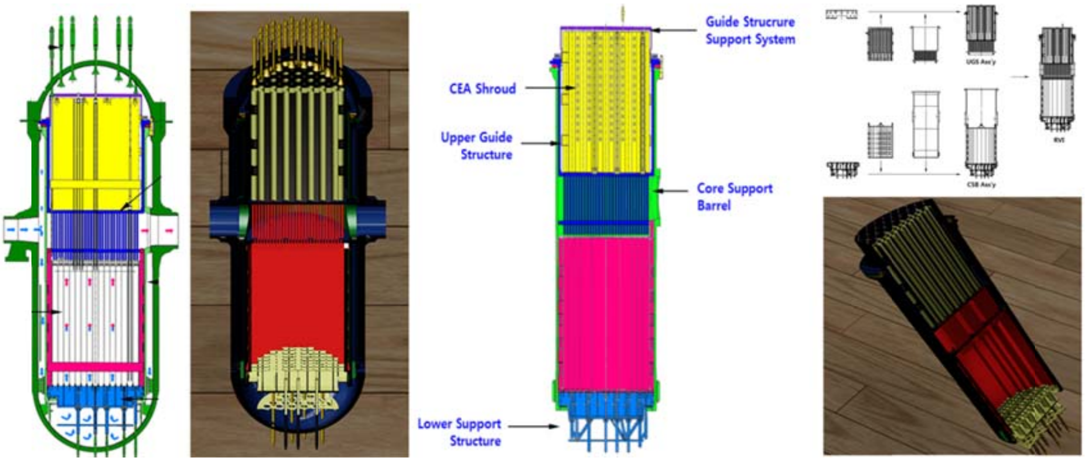


Fig. 1 Schematics and VR snapshots for reactor internals of OPR1000

Table 1 Major dimensions of reactor internals

Component	Height (m)	Radius (m)	Thickness (m)
CSB	9.728	1.828	0.076
UGS	4.924	1.735	0.076
LSS	1.682	1.735	0.084
GSSS	0.914	1.791	0.025

assembly includes the UGS barrel assembly, the control element assembly (CEA) shroud assembly, the guide structure support system (GSSS), the hold-down ring and the heated junction thermocouple shroud assembly. The UGS assembly provides the CEAs a protection from the coolant flow and limits upward motion of the fuel assemblies (Jhung and Hwang 1996).

Since details of the reactor internals are different according to plant types and vendors, in this research, we focused on reactor internals of OPR1000 (Optimized Power Reactor 1000) as shown in Fig. 1. Based on inherent functionalities, the reactor internals can be classified into six subcomponents; CSB, UGS, CSA, LSS, GSSS and CEA shroud. Major dimensions of each component are shown in Table 1.

2.2 CVAP test

The CVAP (Comprehensive Vibration Assessment Program) is a well-known vibration evaluation plan fulfilled during preoperational and initial test conditions by considering the flow-induced vibration at normal and excessive conditions. By applying the CVAP, it is able to check the integrity and stability margins of the reactor internals, and use them as design input data of the same type NPPs.

Prior to commercial operation, one of the foremost means of verifying the integrity of a reactor is achieved through the RVI (Reactor Vessel Internals) CVAP of the RG (Regulatory Guide) 1.20 (USNRC 2007). The RVI CVAP consists of analysis, measurement, and inspection programs that operate in conjunction with verification of the structural integrity, to establish a feasible margin of safety (Chang *et al.* 2013).

Table 2 Hydrodynamic analysis conditions

Temperature (^{o}C)	Pressure (MPa)	Periodic loads		Random loads
		Pump pressure under 20Hz (Pa)	Pump pressure under 480Hz (Pa)	Flow rate of 1A~2B (kg/sec)
290.6	15.3	1172.1	3447.4	2382.3

Table 3 Material properties

Young's modulus (GPa)	Poisson's ratio	Density (kg/m^3)	Damping factor
199.81	0.29	7888.77	0.01

3. Numerical modeling

3.1 Analysis condition

Two kinds of fluid analyses are carried out according to the hydrodynamic loads; one is periodic load by RCP (Reactor Coolant Pump) pulsation and transverse flow swirl deviation and the other is random load caused by the turbulence flow of coolant.

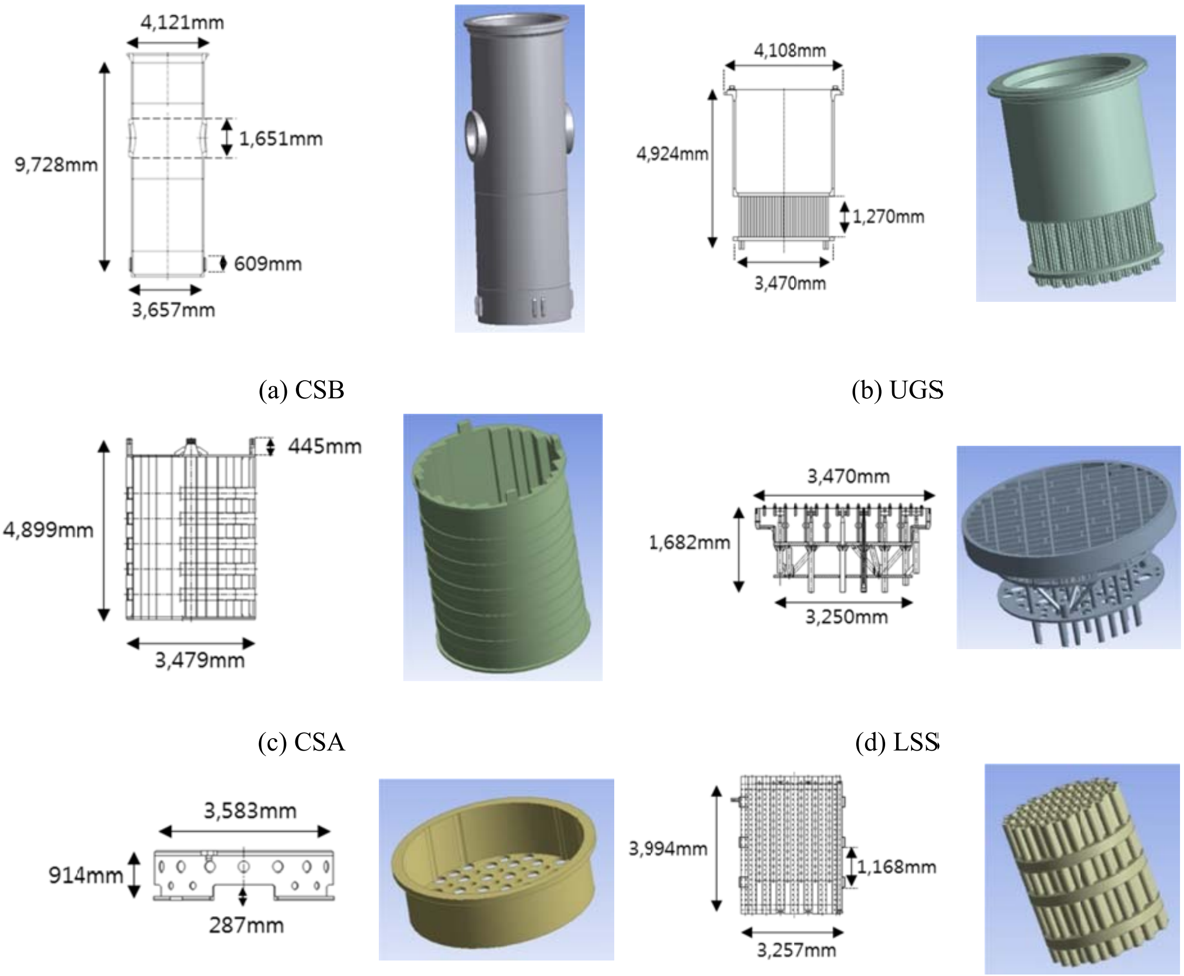
The periodic hydrodynamic load applied to the reactor internals intrinsically is generated at the multiple frequency of pump axial rotation frequency and blade passing frequency. This load was used for acoustic analyses by assuming that it is delivered into the shape of pressure wave independently from flux (Kim *et al.* 2010). The random hydrodynamic load applied to the reactor internals depends on transients as well as the pump operating conditions. This load was used for turbulence analyses by assuming that it involves the highest temperatures and pressures with largest flow rate amounts surrounding the reactor internals.

In this study, a typical CVAP test condition of the KSNP (Korean Standardized Nuclear Power Plant) was retrieved for numerical analyses; 4 RCPs (1A~2B) are operable and details of the hydrodynamic conditions are summarized in Table 2. During the simulation, the cooling water as a medium has the temperature, pressure and density applicable to a typical CVAP test condition. It was assumed that the average density and pressure of cooling water are uniform as summarized in Table 3.

3.2 Analysis model

3-D CAD models were made according to design documents of the KSNP and used for development of an integrated VR prototype. As shown in Fig. 2, all the geometries of subcomponents explained in Section 2.1 were modelled. However, based on preliminary analysis results by the authors, the geometry was optimized to consider numerical analysis capability; one is net modelling of important subcomponents and the other is simplification of the model.

With regard to the former consideration, the GSSS and CEA shroud were eliminated because those were insignificant in fluid flow analyses. In accordance with a previous research, the cooling water moving to the upper head of RPV through UGS was less than 0.1% of total quantity in its flow and has a negligible effect. So, the flow of upper head field was not analyzed and relevant pipes were blocked. With regard to the latter consideration, the simplification process applied only



(a) CSB (b) UGS

(c) CSA (d) LSS

(e) GSSS (f) CEA Shroud

Fig. 2 Geometry and initial CAD models of each subcomponent of reactor internals

to the UGS and LSS. For the UGS, four or three flow channels and pipes were merged into one equivalent flow channel and pipe in proportion to the size while the outermost small flow channels sustained their original shapes. For the LSS, the same philosophy was applied. For other subcomponents, geometries were set to the same with those of the initial CAD models. Fig. 3 shows the final integrated CAD model through the simplification, which was used to generate detailed numerical analysis models.

Fig. 4 represents the mesh of numerical model used for both periodic and random hydrodynamic analyses. It consists of 1,249,818 nodes and 6,398,678 elements having mesh quality of 98%. On the other hand, Fig. 5 represents boundary conditions assigned to the model. Restrictions for each part of reactor internals conformed to actual assembly states, which were similar to those of reference (Kim *et al*. 2009). In case of the RPV, its boundary conditions were set to the basis of reference (Choi *et al*. 2012). Particularly, bottom sides of the cold leg and hot

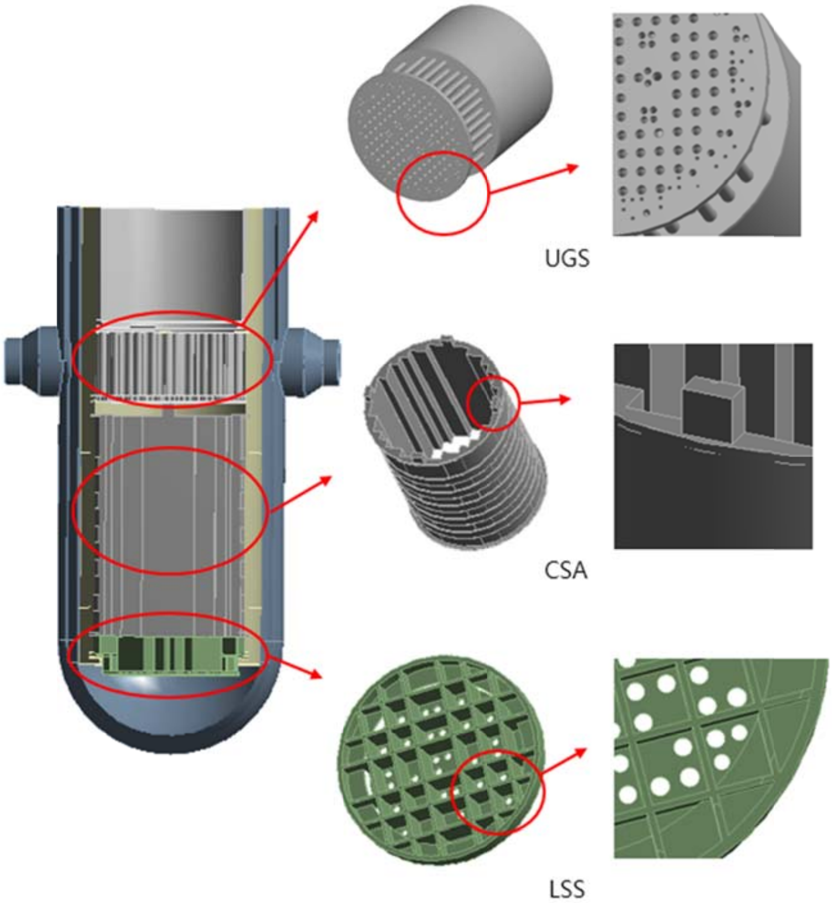


Fig. 3 Integrated CAD model of reactor internals used in the present study

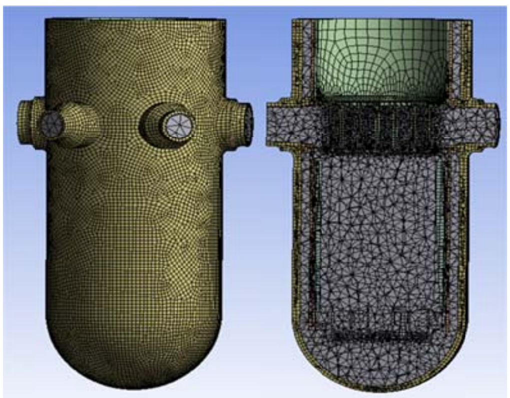


Fig. 4 Mesh of hydrodynamic analysis model including structural region

leg pipes connected to the RPV were fully fixed, and upper head region of the RPV were fixed along axial and tangential directions.

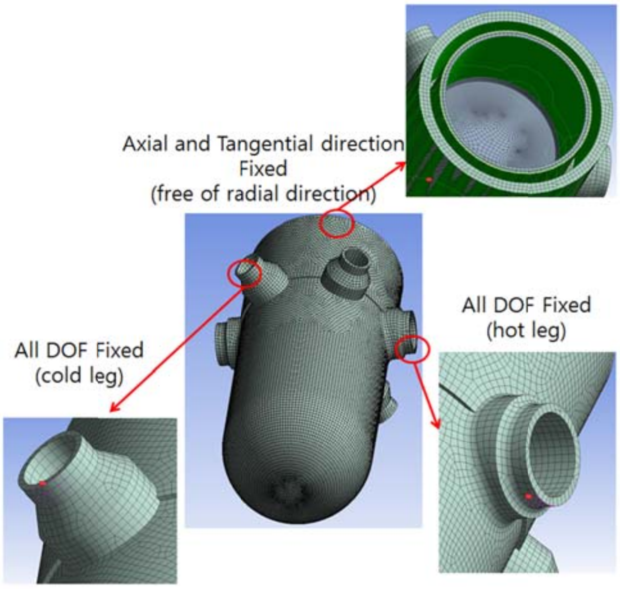
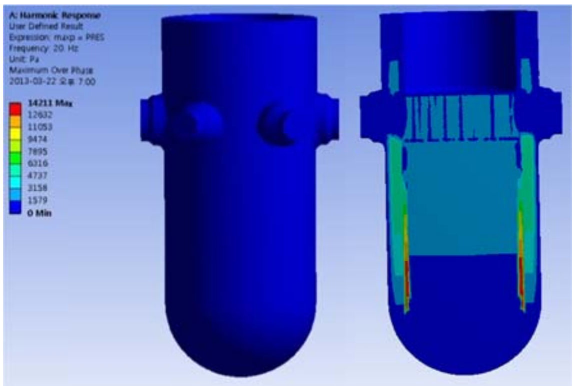
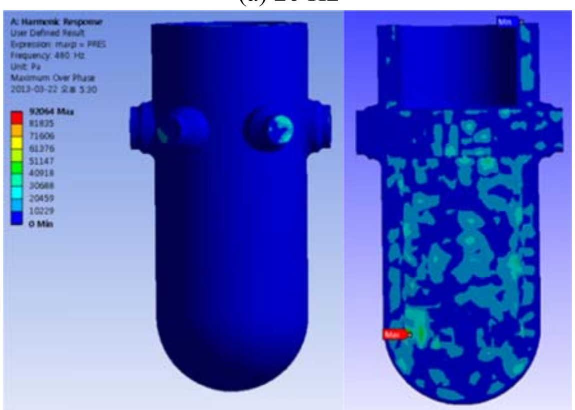


Fig. 5 Boundary conditions



(a) 20 Hz



(b) 480 Hz

Fig. 6 Acoustic pressures distributions obtained from periodic hydrodynamic analyses

4. Numerical analysis and results

4.1 Hydrodynamic analysis based on deterministic forcing functions

With regard to this kind of analyses, we employed 3-D solid elements (Solid 186 in ANSYS library) for the structural region that exhibit quadratic displacement behaviour and acoustic fluid elements (Fluid 30 in ANSYS library) for the fluid medium respectively. The bounding pulsation loads summarized in Table 2 were taken as the minimum and maximum ones from total of five pulsation loads, which were applied to the cooling water-inserted inlet.

Fig. 6 compares resulting acoustic pressure distributions under two RCP pump pulsation conditions. In 20 Hz case, continuously changing distribution was observed along the longitudinal direction beneath the hot and cold leg nozzles. However, in 480 Hz case, impartially spreading distribution was obtained with relatively higher values. According to these quite different acoustic pressure distributions and from the structural integrity assessment point of view, we have to take care of deciding critical locations which were significantly affected by rotor and blade passing frequencies.

4.2 Hydrodynamic analysis based on random forcing functions

In this kind of analyses, we employed two equation turbulent models such as *k-ε* model, *k-ω* model and shear stress transport (SST) model. Among relevant theory to define each turbulent model, differential transport equations were briefly compared (ANSYS 2012); the *k-ε* model assumes that the turbulence viscosity (μ_t) is linked to the turbulence kinetic energy (k) and dissipation rate (ε) via the relation of $\mu_t = C_\mu \rho k^2/\varepsilon$. Values of the k and ε are obtained directly from the following equations.

$$\frac{\partial(\rho k)}{\partial t} + \nabla\cdot(\rho U k) = \nabla\cdot\left[\left(\mu + \frac{\mu_t}{\sigma_k}\right)\nabla k\right] + P_k - \rho\varepsilon \tag{1}$$

$$\frac{\partial(\rho\varepsilon)}{\partial t} + \nabla\cdot(\rho U \varepsilon) = \nabla\cdot\left[\left(\mu + \frac{\mu_t}{\sigma_\varepsilon}\right)\nabla \varepsilon\right] + \frac{\varepsilon}{k}\left(C_{\varepsilon 1} P_k - C_{\varepsilon 2}\rho\varepsilon\right) \tag{2}$$

In the *k-ω* model, the μ_t is linked to the turbulence kinetic energy and frequency (ω) as $\mu_t = \rho k/\omega$. Values of the k and ω are obtained from the following equations.

$$\frac{\partial(\rho k)}{\partial t} + \nabla\cdot(\rho U k) = \nabla\cdot\left[\left(\mu + \frac{\mu_t}{\sigma_k}\right)\nabla k\right] + P_k - \beta^{'}\rho k\omega \tag{3}$$

$$\frac{\partial(\rho\omega)}{\partial t} + \nabla\cdot(\rho U \omega) = \nabla\cdot\left[\left(\mu + \frac{\mu_t}{\sigma_\omega}\right)\nabla \omega\right] + \alpha\frac{\omega}{k}P_k - \beta\rho\omega^2 \tag{4}$$

With regard to the SST model, the above Eq. (3) is sustained but Eq. (4) is changed as below

$$\frac{\partial(\rho k)}{\partial t}+\nabla\cdot(\rho U k)=\nabla\cdot\left[\left(\mu+\frac{\mu_t}{\sigma_{\omega 3}}\right)\nabla\omega\right]+(1-F)2\rho\frac{1}{\sigma_{\omega 2}\omega}\nabla k\nabla\omega+\alpha_3\frac{\omega}{k}P_k-\beta_3\rho\omega^2 \quad (5)$$

where, ρ is the fluid density, U is the velocity vector, μ is the viscosity, μ_t is the turbulent viscosity, σ_k, σ_ε and σ_ω are the turbulent Prandtl numbers, P_{kt} and $P_{\varepsilon t}$ are the thrust force influences and P_k is the turbulence production caused by the viscous forces and F is the blending function. In addition, C_μ, $C_{\varepsilon 1}$, $C_{\varepsilon 2}$, α, β and so on are constants used in the analysis.

To simulate turbulent behaviour, at first, the *k-ε* model was selected because it has been widely used and validated due to relatively simple constituent equation and well convergent algorithm. Excellent results are anticipated provided full turbulence domain supply while poor results may be derived under swirl and vortex flow, separation flow, recirculation and flow conditions having stream curvature effect. The *k-ω* model was selected because it provides more accurate results for boundary-layer flow problems than *k-ε* model does. The last SST model was selected because it is a hybrid of *k-ε* and *k-ω* models. Since this model uses not only *k-ω* model equation at near-wall layer but also *k-ε* model at free stream layer, highly accurate predictions are available for particular problems when transport of turbulent shear stress is prevailing.

Except that applying the average pressure value of 15.3 MPa for the cooling water-discharged outlet as a boundary condition, other analysis conditions were the same with those in the periodic hydrodynamic analyses. Aside the pro et contra of different turbulent models, random hydrodynamic analysis results were compared in Tables 4 and 5. Here, we can find out that the *k-ε* and SST models provide similar distributions especially in velocity fields at several sections as well as pressure and temperature fields although the *k-ω* model shows quite different results. Moreover, since the time needed for the simulation with the *k-ε* and SST models was similar, those can be used as the optimum ones for reactor internals subjected to the random hydrodynamic load.

5. Conclusions

This study was carried out to examine fluid flow of reactor internals subjected to hydrodynamic loads, from which the following conclusions were made.

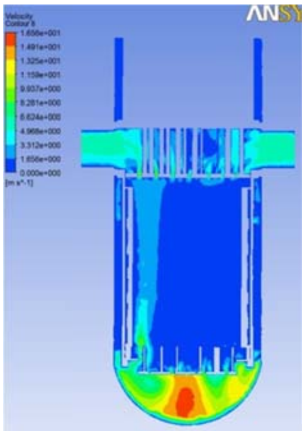
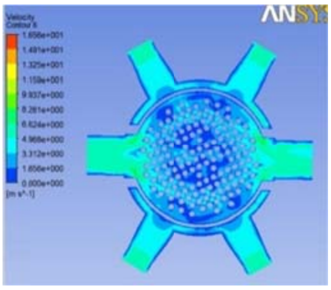
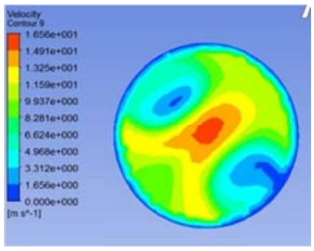
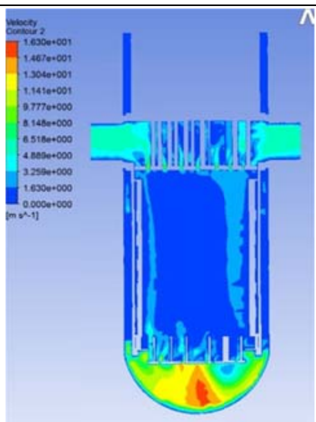
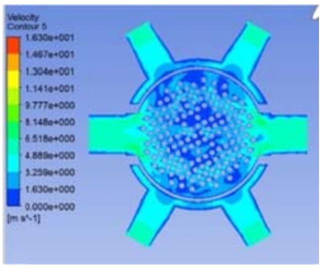
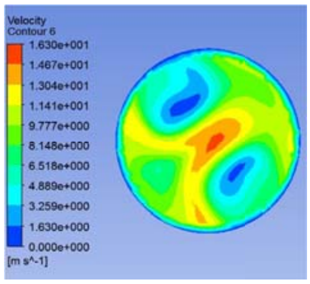
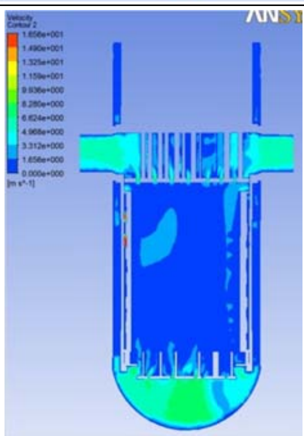
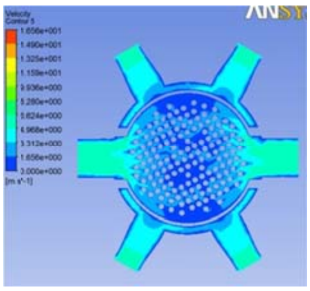
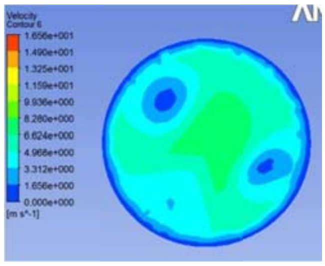
(1) An integrated model consisting of six subcomponents (core support barrel, upper guide structure, core shroud assembly, lower support structure, guide structure support system and control element assembly shroud) was developed by 1,249,818 nodes with mesh quality of 98% and two kinds of numerical analysis methods were established.

(2) The periodic hydrodynamic analysis results showed quite different acoustic pressure distributions under two RCP pump pulsation conditions, which means that rotor and blade passing frequencies can affect critical locations from the structural integrity assessment point of view. Despite of the complexity, resulting stress values were high in the order of CSA, UGS, CSB and LSS in the both conditions.

(3) The random hydrodynamic analysis results showed that the *k-ε* and SST models generate similar distributions especially in velocity as well as pressure and temperature fields but the *k-ω* model showed quite different results. Based on the *k-ε* and SST analyses, resulting stress values were high in the order of UGS, CSB, CSA and LSS.

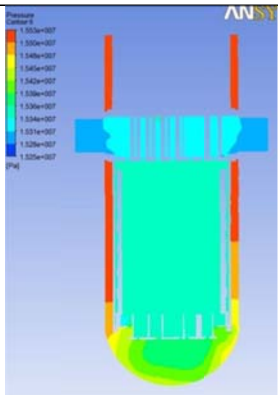
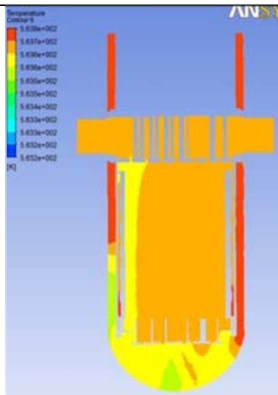
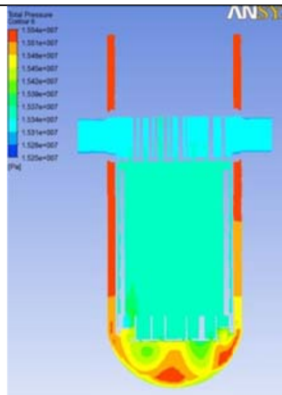
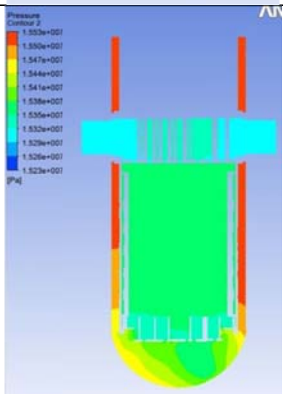
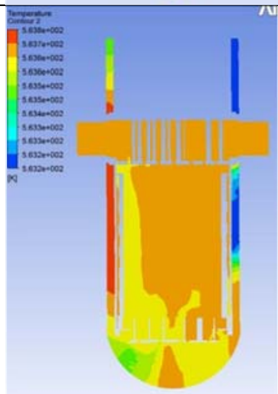
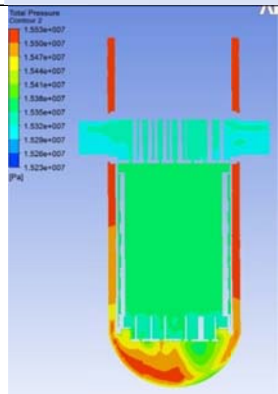
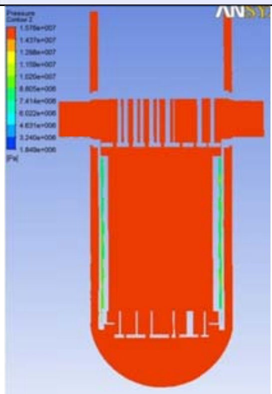
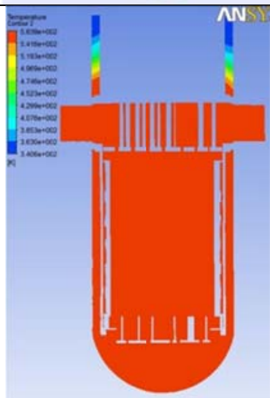
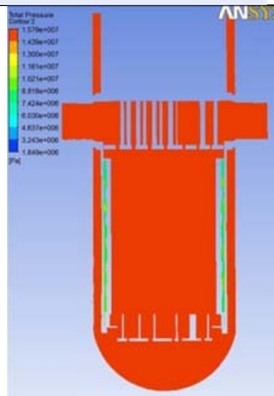
(4) From the view point of structural integrity of the reactor internals, stresses due to the random hydrodynamic loading were relatively higher than those due to the periodic hydrodynamic

Table 4 Comparison of velocity contours at different sections

Turbulent model	Velocity distribution of section 1	Velocity distribution of section 2	Velocity distribution of section 3
SST			
k-ε			
k-ω			

loading. Therefore, UGS and CSB were derived as critical subcomponents for in-depth investigations in the future, and further detailed modeling is being carried out around them.

Table 5 Comparison of pressure, total pressure and temperature contours at section 1

Turbulent model	Pressure	Temperature	Total pressure*
SST			
k-ε			
k-ω			

Note *: Pressure obtained from the periodic load + Pressure obtained from the random load

References

ANSYS Workbench ver.14.0 (2012).
Autodesk inventor professional (2011).
Autodesk showcase professional (2012).

Bae, Y., Kim, Y.I. and Park, C.T. (2013), "CFD analysis of flow distribution at the core inlet of SMART", *Nucl. Eng. Des.*, **258**, 19-25.

Chang, Y.S., Jhung, M.J., Lee, B.S., Kim, H.S. and Hur, N.S. (2013), *Structural Integrity of Nuclear Components*, Hanshouse, Seoul, Korea.

Choi, S.B., Chang, Y.S., Choi, J.B., Kim, Y.J., Jhung, M.J. and Choi, Y.H. (2010), "Fatigue life assessment of reactor coolant system components by using transfer functions of integrated FE model", *Nucl. Eng. Tech.*, **42**(2), 590-599.

Gu, J.Y., Kim, K.H. and Kim, Y.S. (2010), "Development and validation of structural analysis methodology for comprehensive vibration assessment of reactor vessel internals", *Proceedings of the KSME 2010 Fall Annual Meeting*, 950-955.

Hermansky, P. and Krajcovic, M. (2011), "The numerical simulation of the WWER440/V213 reactor pressure vessel internals response to maximum hypothetical large-break loss of coolant accident", *Nucl. Eng. Des.*, **241**(4), 1177-1183.

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

Jhung, M.J., Yu, S.O. and Lim, Y.T. (2011), "Dynamic characteristics of a partially fluid-filled cylindrical shell", *Nucl. Eng. Tech.*, **43**(2), 167-174.

Kao, M.T., Wu, C.Y., Chieng, C.C., Xu, Y., Yuan, K., Dzodzo, M., Conner, M., Beltz, S., Ray, S. and Bissett, T. (2011), "CFD analysis of PWR core top and reactor vessel upper plenum internal subdomain models", *Nucl. Eng. Des.*, **241**(10), 4181-4193.

Kim, K.H., Ko, D.Y. and Kim. Y.S. (2009), "Hydraulic and structural analysis methodology of RVI CVAP in Shin-Kori 4", *Transaction of the Korean Nuclear Society Spring Meeting*, Pyeongchang, Korea.

Kim, K., Euh, D.J., Chu, I.C., Youn, Y.J., Choi, H.S. and Kwon, T.S. (2013), "Experimental study of the APR+ reactor core flow and pressure distributions under 4-pump running conditions", *Nucl. Eng. Des.*, **265**(12), 957-966.

Kim, K.M., Lee, B.I., Lee, D., Cho, H.H., Park, J.S. and Jeong, K.H. (2012), "Pump-induced pulsating pressure distributions in a system-integrated modular reactor", *Nucl. Eng. Des.*, **248**, 216-225.

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*, Jeju, Korea.

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", *Tran. Korean Soc. Noise Vib. Eng.*, **21**(8), 708-713.

Ko, D.Y. and Kim, K.H. (2013), "Structural analysis of CSB and LSS for APR1400 RVI CVAP", *Nucl. Eng. Des.*, **261**, 76-84.

Lee, B.I., Kim, K.M., Lee, D.H., Cho, H.H., Jeong, K.H. and Park, J.S. (2012), "Measurement of flow-induced pressure variation in a scale-down SMART model", *Nucl. Eng. Des.*, **253**, 50-59.

Park, S.Y. and Cho, J.R. (2012), "Adaptive fluid-structure interaction simulation of large-scale complex liquid containment with two-phase flow", *Struct. Eng. Mech.*, **41**(4), 559-573.

USNRC (2007), "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing", Regulatory Guide 1.20, Rev.0.

USNRC (2010), "Generic Aging Lessons Learned (GALL) Report", NUREG-1801, Rev.2.

Tsuji, N., Nakano, M., Takada, E., Tokuhara, K., Ohashi, K., Okamoto, F., Tazawa, Y., Inaba, Y. and Tachibana, Y. (2014), "Study of the applicability of CFD calculation for HTTR reactor", *Nucl. Eng. Des.*, **271**, 564-568.

Wu, C.Y., Ferng, Y.M., Chieng, C.C. and Kang, Z.C. (2012), "CFD analysis for full vessel upper plenum in Maanshan nuclear power plant", *Nucl. Eng. Des.*, **253**, 285-293.

Xu, Y., Conner, M., Yuan, K., Dzodzo, M.B., Karoutas, Z., Beltz, S.A., Ray, S., Bissett, T.A., Chieng, C.C, Kao, M.T. and Wu, C.Y. (2012), "Study of impact of the AP1000 reactor vessel upper internals design on fuel performance", *Nucl. Eng. Des.*, **252**, 128-134.