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Hydraulic and Structural Analysis for APR1400 Reactor Vessel Internals against Hydraulic Load Induced by Turbulence

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Abstract : The structural integrity assessment of APR1400 (Advanced Power Reactor 1400) reactor vessel internals has been being performed referring the US Nuclear Regulatory Commission regulatory guide 1.20 comprehensive vibration assessment program prior to commercial operation. The program is composed of a hydraulic and structural analysis, a vibration measurement, and an inspection. This paper describes the hydraulic and structural analysis on the reactor vessel internals due to hydraulic loads caused by the turbulence of reactor coolant. Three-dimensional models were built for the hydraulic and structural analysis and then hydraulic loads and structural responses were predicted for five analysis cases with CFX and ANSYS respectively. The structural responses show that the APR1400 reactor vessel internals have sufficient structural integrity in comparison with the acceptance criteria.

Key words : CVAP, reactor vessel internals, NRC RG1.20, analysis program, turbulence

1. Introduction

The US NRC (National Regulatory Committee) Regulatory Guide 1.20 requires a comprehensive vibration assessment program (CVAP) to verify the structural integrity of reactor vessel internals (RVI) against flow-induced vibration during preoperational and initial startup testing. The CVAP consists of a vibration and stress analysis, a vibration measurement program, and an inspection program [1]. The goal of the vibration and stress analysis is to theoretically verify the structural integrity of RVI and to provide a basis for selecting the locations monitored in the vibration measurement and inspection program.

The vibration and stress analysis consist of a hydraulic analysis to estimate flow-induced dynamic loads and a structural analysis to predict structural responses to the dynamic loads. There are two types of flow-induced dynamic loads; deterministic hydraulic loads caused by pump pulsation and vortex shedding due to cross flow, and random hydraulic loads caused by turbulent flow [2,3].

In Combustion Engineering (CE) nuclear power plants, such as Yonggwang Unit 4, Ulchin Unit 5 and 6, and Shin-Kori Unit 1 in Korea, random hydraulic analyses were performed using the CVAP data of Palo Verde Unit 1 and Yonggwang Unit 4. The structural response analyses included the use of simple finite element models or multiple-degree-of-freedom lumped mass-beam element models, depending on the complexity of the RVI and the characteristics of the hydraulic loads [3~6].

These analysis methods used the CVAP measurement data of previous plants or were excessively conservative, since simple models were used due to limited computer resources. In this study, to reduce such conservatism and achieve more realistic results, three-dimensional (3-D) fluid and structural models were built to perform the random hydraulic analysis for Advanced Power Reactor (APR) 1400 RVI CVAP. This paper also covers the structural integrity of RVI to turbulent flow in comparison with the acceptance criteria defined according to the American Society of Mechanical Engineers (ASME) code.

2. Analysis Method

Fig. 1 represents the method to predict the random hydraulic load induced by turbulence and the structural responses of the RVI [7~9]. The flow in the upper guide structure (UGS) is less than 3% of total coolant flow in the reactor vessel and the structure of the inner barrel assembly (IBA) is too complicated to simulate.

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Thus, the nearly stagnant flow in the UGS is excluded from the scope of the analysis in the reactor vessel. The 3-D CAD solid models of the RVI are built in accordance with the RVI's design documents, and flow field models in CFX are generated with a subtraction Boolean operation. Turbulence in the reactor vessel is simulated by a shear stress transport (SST) turbulence model for a steady state analysis and a detached eddy simulation (DES) turbulence model for a transient analysis. The flow rates at which coolant is discharged by reactor coolant pumps (RCP) are input in the inlet nozzles as boundary conditions. Because of the random nature of turbulence, a statistical method is used to define both the magnitude and frequency of the turbulence in the form of power spectral density (PSD), which is also used for the structural response analysis.

The 3-D CAD solid models used for the hydraulic models are also utilized for the structural analysis, and structural analysis models are generated with the SOLID186 element of ANSYS. Since the reactor vessel internals are submerged in coolant, an added mass is calculated for each internal according to ASME B&PV Section III [10]. Natural frequencies and natural modes are calculated with the block Lanczos method, which is commonly used in commercial structural analysis programs. The spectrum analysis is used for the structural response analysis. The range of frequency for the analysis is 0 to 500 Hz and the scale of the analysis results is 3-sigma.

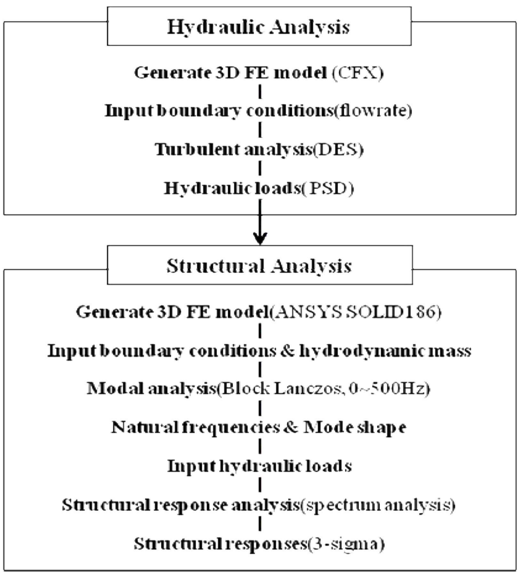

Fig. 1. Method of the hydraulic and structural analysis for RVI.

3. Hydraulic Analysis

3.1 Hydraulic Analysis Model and Analysis Conditions

Hydraulic models were defined by either tetrahedral or hexahedral mesh in CFX to analyze the flow field of the APR1400 reactor (Fig. 2). The flow field included inlet nozzle (cold-leg), downcomer between reactor vessel and core support barrel (CSB), lower support structure (LSS), core, CEA (control element assembly) guide tube bank under the UGS assembly, and outlet nozzle (hot-leg). However, since the measurement and inspection program of the RVI CVAP are performed during the pre-core hot functional test (HFT), fuel assembly was not included in the flow field. For the transient analysis, three 3-D hydraulic models (① inlet to downcomer, ② downcomer to core, ③ core to outlet) were built due to computer hardware. In addition, a full 3-D hydraulic model, which includes the entire flow field, was generated to estimate the reference pressures of those transient analysis models.

Table 1 shows temperature, pressure, density, and viscosity of coolant used in the analysis. Table 2 shows the flow rates of the reactor coolant pumps for the five analysis conditions. The negative values represent back flow rates when RCP is not operating.

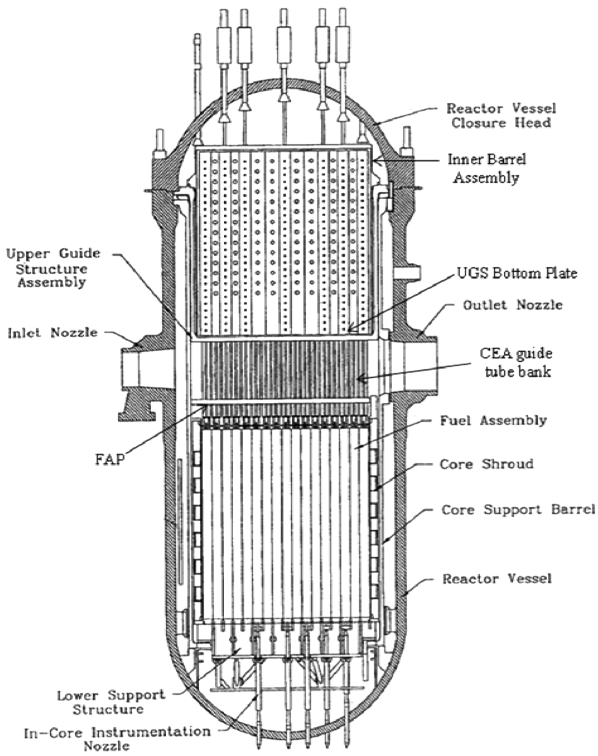

Fig. 2. Reactor vessel assembly of APR1400.

Table 1. Properties of Reactor Coolant

Temperature	555 °F
Pressure	2219 psi
Density	0.02696 lb/ft^3
Viscosity	1.336×10-8 lbf s/in^2

Table 2. Operation conditions of reactor coolant pump

Case No.	RCP(lb/s)			
	Loop 1		Loop 2	
	1A	1B	2A	2B
1	-1,171	-1,171	18,435	-3,265
2	-2,480	-2,480	15,529	15,529
3	18,049	-4,010	18,049	-4,010
4	17,333	-4,960	14,771	14,771
5	11,579	11,579	11,579	11,579

3.2 Results of hydraulic analysis

Velocity contour predicted by the steady state analysis for the full model of case #1 is shown in Fig. 3. Coolant enters RCP 2A through the inlet nozzle and then flows through downcomer into the LSS region. The coolant is mixed in the LSS region, and the mixed coolant flows to the outlet nozzle of loop 2, operated by RCP 2A, passing the core region. The red and yellow region on the right side of Fig. 3 is the outlet nozzle of loop 2.

Fig. 4 describes the PSD of pressure fluctuation on the UGS bottom plate for case #1, which is divided into eight surfaces. The higher values of PSD fall within the relatively low frequency range, but the mean of PSD gradually converges as the frequency increases.

In order to input PSD into the structural response analysis, the surface of a component is divided into several areas, in consideration of the pressure distribution

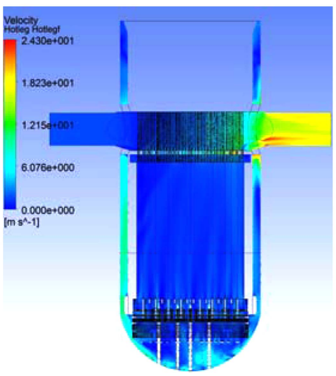

Fig. 3. Velocity contour of steady state under case #1.

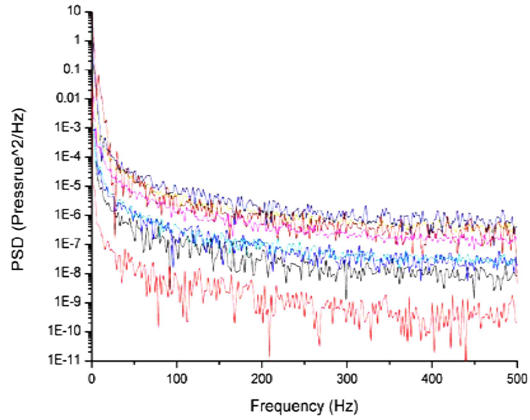

Fig. 4. Pressure PSD on the UGS bottom plate under case #1.

of the component as calculated by steady state analysis for the full model. However, the PSD of the outside surface of the UGS bottom plate is conservatively input into the entire surface of IBA, since the hydraulic loads in IBA are not predicted.

4. Structural Analysis

4.1 Structural analysis model and analysis conditions

The reactor vessel internals in Fig. 2 were categorized into four components: CSB, LSS, UGS Assembly (which includes UGS, CEA guide tube bank and FAP), and IBA. Four structural models were generated for the four components using SOLID186 element in ANSYS.

The material of the RVI is austenitic stainless steel and the properties of the material are shown in Table 3. The damping factor used in the structural analysis is 0.01%. To consider the effect of the coolant surrounding the RVI, the hydrodynamic mass was calculated according to ASME code, and the mass was added with ANSYS SURF154 element. The constraints of the models were chosen based on the configuration of assembling the internals and the dynamic conditions of the internals.

4.2 Results of structural analysis and assessment of structural integrity

Table 4 shows the first natural frequency of each

Table 3. Properties of RVI material

Elastic Modulus	29,000 ksi
Poisson’s Ratio	0.29
Density	0.289 lb/in^3

Table 4. First natural frequencies of RVI

Component	First natural frequency (Hz)
CSB	12.81
LSS	121.76
UGS Assembly	18.86
IBA	35.38

component. The more complicated the component is in the order of LSS, IBA, UGS, CSB, the higher the natural frequency and stiffness the component has.

Stress contours of the components under case #1 are shown in Figs. 5 – 8. For this case, relatively higher stress appears in the CSB. The peak stress of each component is shown in Table 5. The highest peak stress is shown in the CSB, since it also has the lowest natural frequency. The peak stress for each component appears in the region between the CSB flange and the outlet nozzle in the CSB; in the welding region between the in-core instrumentation (ICI) nozzles and the support structure in the LSS; in the welding region between the CEA guide tubes and the FAP in UGS Assembly; and in the welding region between the outermost webs and the IBA inner wall in the IBA.

For the analyzed cases, the highest peak stress of the CSB occurs when three reactor coolant pumps are operating. The structural responses in the LSS, UGS Assembly, and IBA vary depending on the operating conditions of the RCP, but their values are much lower than that of the CSB.

The acceptance criteria of the structural vibration response are used as the criteria of the integrity assessment for the RVI. The stress acceptance criterion of the structural vibration response is a third of 13.6ksi which is the endurance limit equivalent to 10[11] cycles in

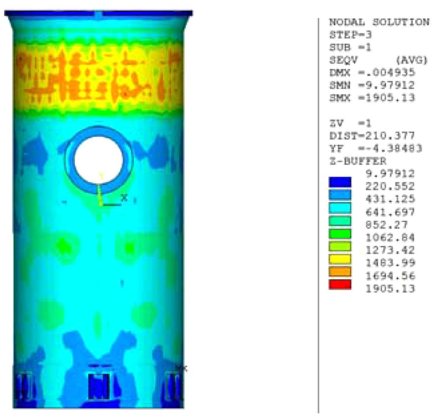

Fig. 5. Stress contour of CSB under case #1.

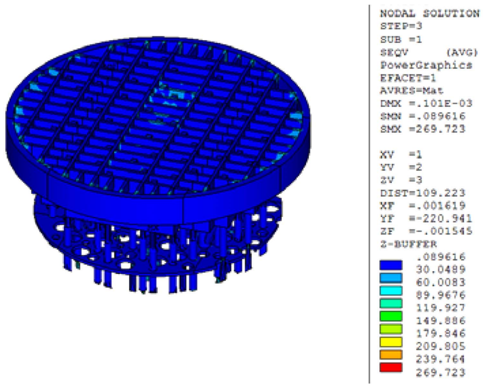

Fig. 6. Stress contour of LSS under case #1.

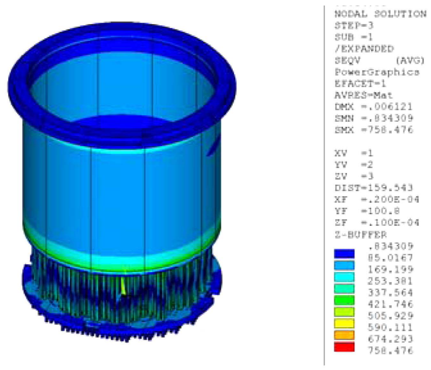

Fig. 7. Stress contour of UGS Assembly under case #1.

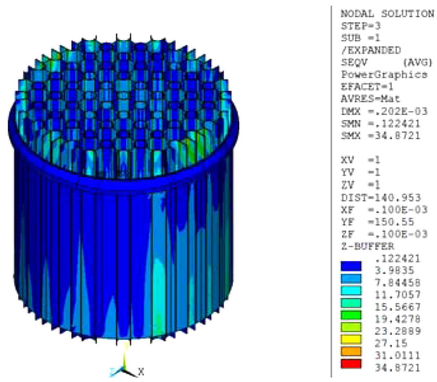

Fig. 8. Stress contour of IBA under case #1.

ASME B&PV Section III, Division 1, Appendix I, Fig. I-9.2 Design Fatigue Curves [12]. As shown in Table 6, the fatigue margin for each component is found by dividing the acceptance criteria by the highest peak stress of each component among the analysis cases. The lowest fatigue margin of the RVI is 1.85, which means

Table 5. Peak stress(psi) of components

Case No.	CSB	LSS	UGS Assembly	IBA
1	1,480	60.1	169	7.84
2	1,530	11.4	498	257
3	1,280	21.0	113	95.8
4	2,450	22.5	113	25.5
5	1,540	33.7	358	298

Table 6. Peak Stress and Fatigue Margin

Component	Peak stress (psi)	Acceptance criteria (psi)	Fatigue margin
CSB	2,450	4,533	1.85
LSS	60.1	4,533	75.4
UGS Assembly	498	4,533	9.10
IBA	298	4,533	15.2

the RVI has good structural integrity against turbulent flow.

5. Conclusions

In this paper, in order to predict the hydraulic loads induced by turbulent flow in the reactor vessel for APR1400 reactor vessel internals comprehensive vibration assessment program, 3-dimensional hydraulic models were generated and the loads were estimated with CFX. Also, four 3-dimensional structural models were built to predict the structural responses of the reactor vessel internals to the hydraulic loads and the responses were predicted with ANSYS. The peak stress occurred in the core support barrel and the fatigue margin to the stress acceptance criterion was 1.85. Thus, the structural safety of the reactor vessel internals to turbulent flow of coolant was assured.

In future works, these structural responses will be summated with structural responses to deterministic hydraulic loads caused by reactor coolant pumps. The structural integrity of the summated responses will then be theoretically assessed and used to determine where to install measurement instruments for the vibration measurement program of Shin-kori Unit 4. The results of the vibration and stress analysis and the structural integrity of the APR1400 reactor vessel internals will be further verified by the vibration measurement program of the Shin-kori Unit 4 reactor vessel internals comprehensive vibration assessment program, which will be performed during pre-core hot functional test in 2013.

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