Journal of KSNVE, Vol. 7, No. 1, pp. 33~41, 1997.

(Original Paper)

The Effect of Seismic Level Increase on the Reactor Vessel Internals and Fuel Assemblies for the Korean Standard Nuclear Power Plant

지진레벨의 증가가 한국표준형 원자력발전소의 원자로 내부구조물 및 핵연료집합체에 미치는 영향

M. J. Jhung*, Y. W. Park** and J. B. Lee***

정 명 조 · 박 윤 원 · 이 정 배

(Received August 9, 1996 ; Accepted November 5, 1996)

Key Words : Reactor Vessel Internals (원자로 내부구조물), Fuel Assembly (핵연료집합체), Seismic Level (지진레벨), Earthquake (지진)

ABSTRACT

To cover a range of possible site conditions where the Korean standard nuclear power plant may be constructed, a range of generic site conditions is selected for geologic and seismologic evaluation. To envelop other Asian countries as well as the Korean peninsula, there is an attempt to increase the seismic level to 0.3g ground motions for the safe shutdown earthquake. The dynamic analyses of the reactor vessel internals and fuel assemblies are performed for the increased motions and the effect of seismic level on the response is investigated. Also the nonlinear response characteristics are discussed by comparing the loads between operating basis earthquake and safe shutdown earthquake excitations. The design adequacy of the reactor vessel internals and fuel assemblies for the increased seismic level is addressed.

요 약

경수로형 원자력발전소 표준화작업의 일환으로 만들어진 한국표준형 원자력발전소는 그 건설부지를 한반도 뿐만 아니라 인접 아시아국가의 여러 곳을 목표로 하고 있으며 이와 관련하여 안전정지지진의 레벨을 0.3g로 증가시키려는 시도가 계획되고 있다. 본 연구에서는 이와 같은 지진레벨 증가가 기존의 0.2g로 설계된 원자로 내부구조물과 핵연료집합체에 미치는 영향을 평가하였다. 운전기준지진 및 안전정지지진의 응답을 비교함으로써 비선형 응답특성을 조사하였고 한국표준형 원자력발전소의 원자로 내부구조물 및 핵연료집합체의 설계 타당성에 대하여 언급하였다.

1. Introduction

As an attempt to standardize a pressurized water reactor type nuclear power plant in Korea, the Korean standard nuclear power plant (KSNPP) is designed demonstrating the compliance of the standard design with all current regulations for existing plants as well as the guidelines for future plants outlined in the US Nuclear Regulatory Commission's severe accident policy statement.

The standard design contains most of the design

*정회원, 한국원자력안전기술원 기계설비그룹
**한국원자력안전기술원 기술개발그룹
***한국원자력안전기술원 기계설비그룹

features of the Yonggwang nuclear power plant units 3 and 4 which began their commercial operations in 1995 and 1996, and includes a variety of engineering and operational improvements. Specifically, the Electric Power Research Institute's advanced light water reactor requirement document has been used as a guide for utility requirements regarding plant design.

Since the standard design is based on the assumed site-related parameters which are selected to envelop most potential nuclear power plant sites in other Asian countries as well as Korea where the KSNPP may be constructed, the peak ground accelerations for the safe shutdown earthquake (SSE) are considered to be 0.3g and 0.2g for the horizontal and vertical directions, respectively.

In this study, the dynamic analyses of the reactor vessel internals and fuel assemblies are performed for the increased seismic level using the reactor vessel (RV) motions used for the Ulchin nuclear power plant units 3 and 4 design which will be the first KSNPP. The effect of seismic level increase on the response is investigated and the nonlinear response characteristics are discussed by comparing the loads between operating basis earthquake (OBE) and safe shutdown earthquake excitations. Also the design adequacy of the current reactor vessel internals and fuel assemblies which are based on the 0.2 g ground motion is addressed for the increased seismic level.

2. Korean Standard Nuclear Power Plant

The KSNPP containment building is a prestressed, reinforced concrete structure in the shape of a cylinder with a hemispherical dome and a flat foundation slab. The cylindrical portion of the containment structure is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The interior surface of the containment shell is steel-lined for leaktightness. A protective layer of concrete covers the portion of the linear over the foundation slab. The containment structure concrete provides biological shielding for normal and accident conditions.

The nuclear steam supply system (NSSS) generates approximately 2825 MWt, producing saturated steam. The NSSS contains two independent primary coolant loops, each of which has two reactor coolant pumps, a steam generator, a 42-inch Inside Diameter (ID) outlet pipe and two 30-inch ID inlet pipes. An electrically heated pressurizer is connected to one of the loops, and safety injection lines are connected to each of the four cold legs and the two hot legs. Pressurized water is circulated by means of electric-motor-driven, single-stage, centrifugal reactor coolant pumps. Reactor coolant flows downward between the reactor vessel shell and the core support barrel, upward through the reactor core, through the hot leg piping, through the tube side of the vertical U-tube steam generators, and back to the reactor coolant pumps. The saturated steam produced in the steam generators is passed to the turbine.

Automatic protection systems, control systems, and interlocks are provided, along with the administrative controls of the specific site, to assure safe

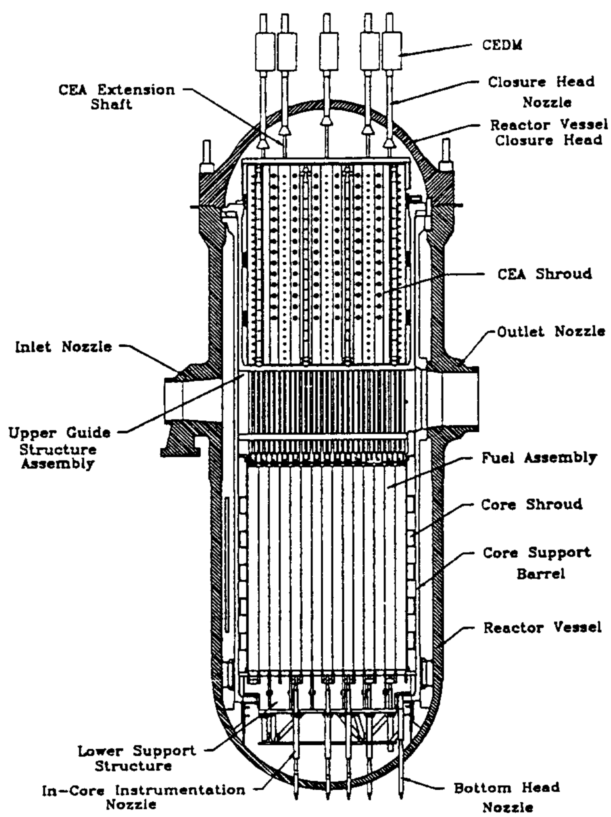


Fig. 1 Schematic diagram of reactor vessel internals.

operation of the plant. Sufficient instrumentation and controls are supplied to provide manual operation as a normal backup control mode on all automatic systems.

The reactor vessel internals are designed to support the reactor core, maintain the core in a coolable array, and guide the control element assemblies (CEAs) into the top of the core, and to constrain and protect the CEAs from coolant flow.

The components of the reactor vessel internals are divided into two major parts consisting of the core support barrel (CSB) assembly and the upper guide structure (UGS) assembly. The flow skirt, although functioning as an integral part of the coolant flow path, is separate from the internals and is affixed to the bottom head of the reactor vessel. The arrangement of these components is shown in Fig. 1.

3. Model Development

3.1 Coupled Internals and Core Model

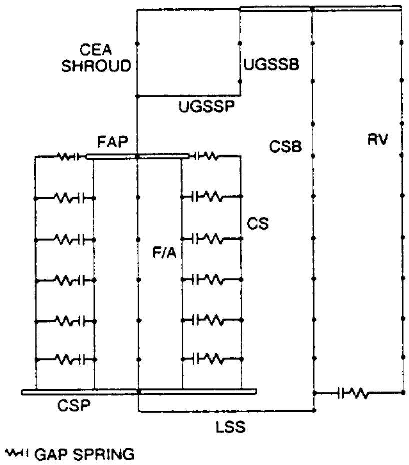
The mathematical model of the internals consists of lumped masses and elastic beam elements to represent the beam-like behavior of the internals, and nonlinear elements to simulate the effects of gaps between components. Typical component gaps represented by nonlinear elements in the horizontal direction are the core support barrel, pressure vessel snubber gap and core shroud guide lug gap. The gaps between the core shroud and core support barrel or the core support plate and core support barrel are sufficiently large that no contacting occurs. For the vertical direction, the gap between guide tube and upper end fitting of the fuel assembly is represented by nonlinear element.

At appropriate locations within the internals and core, nodes are chosen to lump the weights of the structure. The criterion for choosing the number and location of mass points is to provide for accurate representation of the dynamically significant modes of vibration for each of the components. For the beam element connecting two nodes in the horizontal model, properties are calculated for moment of inertia, cross-sectional area, effective shear area, stiffness and length. For the vertical stiffness, a well known formula $K=AE/L$ is used where K, A, E and L are axial stiffness (lb/in), cross-sectional area (in^2), Young's modulus (psi) and length of segment (in), respectively.

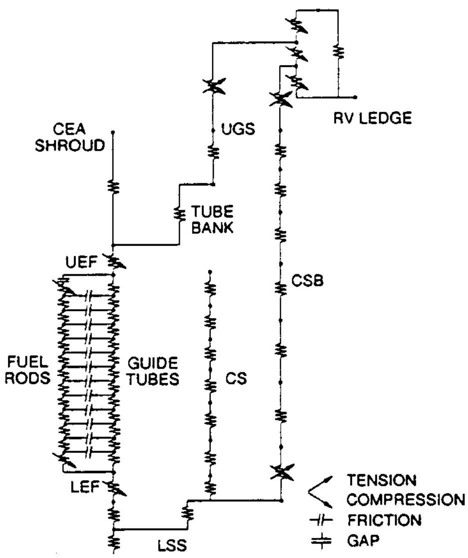
Stiffnesses for the complex structures such as flanges, snubber, hold-down ring and guide tubes are determined by finite element analyses. Unit deflections and rotations are applied and the resulting reaction forces are calculated. These results are then used to derive the equivalent member properties for the structure.

Core model in the coupled internals and core horizontal analysis accounts for the total number of fuel assemblies in the core. Thus, the effect of the entire fuel assemblies on core plate motions is included in the model. The core is modeled by subdividing it into several fuel assembly groupings and choosing stiffness values to adequately characterize its beam response and contacting under dynamic loading[1]. Fig. 2 (a) shows a core region with 3 fuel assembly groupings. The outside groupings represent the peripheral and adjacent row of fuel assemblies in the core. The center grouping is made up of the remaining fuel assemblies. To simulate the nonlinear motion of the fuel, nonlinear spring couplings are used to connect corresponding nodes to the fuel assemblies and core shroud. Incorporated into these nonlinear springs is the spacer grid impact stiffness derived from test results. By simulating actual gaps between the peripheral fuel assemblies and the core shroud, the effects of impacting that can be fed back through the core shroud and alter the motion of the core plates are included in the model.

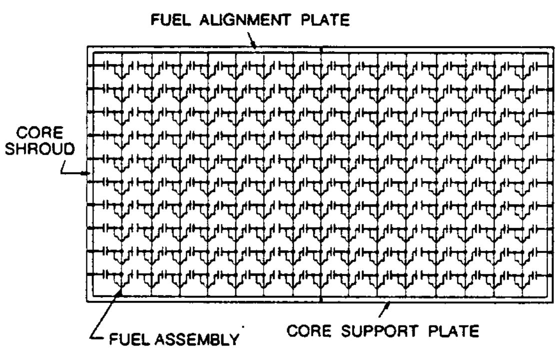
The stiffness properties of each fuel grouping are determined by combining the individual fuel assembly stiffnesses added in parallel. Individual fuel assembly is modeled with beam elements to represent the stiffness between mass points and rotational springs at each end to simulate the end fixity existing at the top and bottom of the core. The values used for stiffness and end fixity are based on a parametric study in which analytic predictions are corrected with fuel assembly static and dynamic test data. The fuel assembly parameters used in the coupled internals and core model are basically the



(a) Horizontal model of reactor vessel internals



(b) Vertical model of reactor vessel internals



(c) Detailed horizontal core model of 15 row fuel assemblies

Fig. 2 Lumped mass models of reactor vessel internals and fuel assemblies.

same as those used in the detailed core model.

The core in the vertical direction is modelled by grouping all fuel assemblies into a single grouping which includes the weight of the entire core. The vertical stiffness properties of the grouping are obtained by combining the individual stiffnesses of all fuel assemblies in the core as springs in parallel. The fuel assembly grouping is subdivided into fuel rods and guide tubes with slip stick friction elements representing the connectivity between the two single stick models. This connectivity represents the friction force between the fuel rods and spacer grid arches and tabs. Both static and dynamic friction values are used[2]. The fuel rods do not slip until the value of static friction is exceeded. When slipping of the fuel rods occurs, the resisting force is equal to the dynamic friction force. Friction values are determined from tests by measuring the forces required to withdraw fuel rods from the bundle.

3.2 Detailed Horizontal Core Model

In the detailed core model, the fuel assemblies are modeled as uniform beams. Lumped masses are included at spacer grid locations to represent the significant modes of vibration of the fuel and to account for possible spacer grid impacting. Nonlinear spring couplings are used to simulate the gaps in the core. Each spacer grid is characterized by the dual load path model which represents the load paths associated with both one-sided and through-grid impacts. One-sided loads are the loads experienced by one side of a grid when it impacts on another grid or the core shroud. Through-grid loads are the loads developed through grid loadings on a spacer grid.

The fuel analytical model was constructed by calculating nodal properties for corresponding locations based on the weight distribution data. The dynamic characteristics of the fuel bundle including natural frequency and damping were also determined from the test data. The analytical model of the fuel bundle was modified to include dynamic effects by adjusting the bundle stiffness to obtain the proper natural frequency and prescribing the damping as a percentage of critical damping[3].

Hydrodynamic (diagonal coupling coefficients) mass was added to the structural mass to obtain the proper natural frequency in water[4,5]. The off-diagonal coupling terms are not considered in the core model, that is, hydraulic coupling between the fuel assemblies is neglected. This was justified by water loop tests[6], which indicate that the natural frequency drop can be accounted for by added masses corresponding to the displaced liquid, meaning that a fuel assembly in a channel does not behave in a significantly different manner as a fuel assembly in an infinite fluid. Physically this means that without a wrapper tube, the fluid can flow from one side of the assembly to the other, across the fuel assembly rather than around it.

The spacer grid model was developed considering impacting of adjacent fuel assemblies or peripheral assemblies and the core shroud. If two fuel assemblies hit another or if one assembly strikes the core shroud, then the spacer grids are loaded on only one force. This type of impact has been called a one-sided impact. The second impact type is called a through-grid impact because the impact force is applied simultaneously to opposite faces of the spacer grid. For example, a through-grid impact occurs when one fuel assembly is lying against the core shroud and a second assembly hits it[7].

Typical coupled internals and core models in the horizontal and vertical directions and detailed core model of fifteen row fuel assemblies are shown in Fig. 2. The actual arrangement and detail in the model may vary with the function of plant design, and the magnitude and nature of the excitation.

4. Analysis

4.1 Input Excitations

The forcing function to the horizontal model consists of acceleration time histories at the RV flange and snubber elevations determined from the reactor coolant system analysis. The reactor vessel is so stiff comparing with internals components that its local effect is negligible. Therefore, only translational accelerations on the RV between the flange and snubbers are computed by linear interpolation and are input into the model. These translational accelerations along the vessel are required for the calculation of hydrodynamic forces between CSB and RV annulus. The input excitations to the vertical model consist of RV flange motion only which is determined from the reactor coolant system analysis.

The acceleration time histories of RV flange and snubber and its corresponding spectra are shown in Fig. 3. The maximum accelerations in the east-west direction for the 0.3g ground motion are 349.8 in/sec^2 at 6.043 seconds and 217.4 in/sec^2 at 11.803 seconds for the RV flange and snubber elevations, respectively.

The input excitations to the detailed core model consist of the translational and angular motions of the core plates and the translational motion of the core shroud. The core shroud is so stiff comparing with fuel assembly that its local effect is negligible. Therefore, only the translational component of the core shroud is used. The input motions are obtained from a seismic analysis of a coupled internals and core model which has a much less detailed representation of the core.

Figure 4 shows the response spectra of the fuel alignment plate (FAP) and the core support plate (CSP). The zero period accelerations (ZPAs) in the east-west direction are 5.122g and 5.590g for FAP and CSP, respectively. The core plate motions are amplified by 5.0 and 6.2 times from the reactor vessel motions in the north-south and east-west directions, respectively.

4.2 Dynamic Response

The response of the reactor vessel internals is computed by the SHOCK code[8], which solves for the response of the structures represented by lumped mass and spring systems under a variety of loadings. This is done by numerically solving the differential equations of motion for an N degree of freedom system using the Runge-Kutta-Gill technique. The equation of motion can represent an axially responding system or a horizontally responding system i.e., an axial motion or a coupled horizontal and rotational motion. The code is designed to handle a

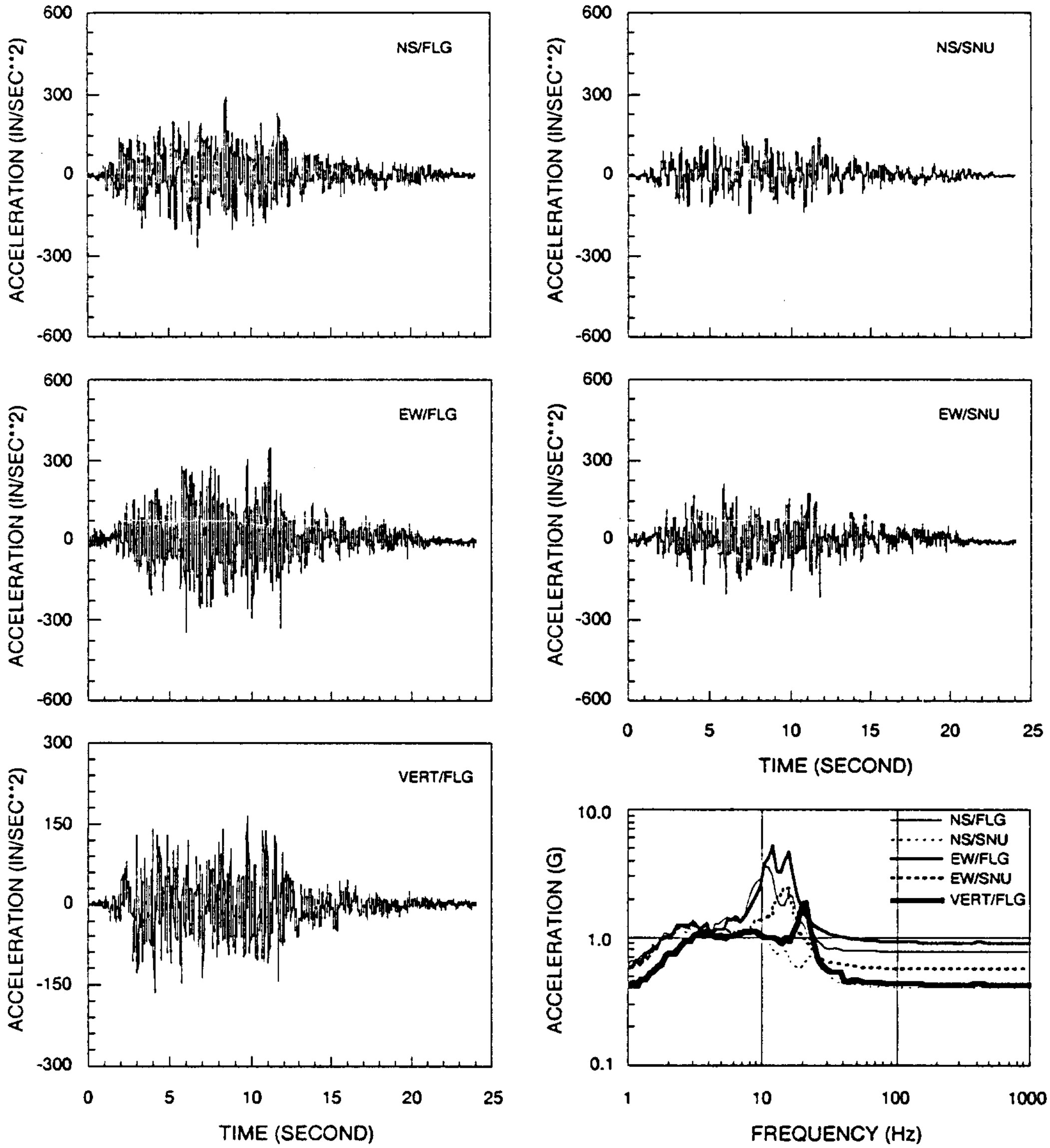


Fig. 3 Reactor vessel motions for 0.3g ground motion.

large number of options for describing load environments and includes such transient conditions as time-dependent forces and moments, initial displacements and rotations, and initial velocities. Options are also available for describing steady-state loads, preloads, accelerations, gaps, nonlinear elements, hydrodynamic mass, viscous damping, friction, and hysteresis.

Equilibrium conditions, prior to the application of the vertical dynamic transient conditions, are established by determining the static displacements associated with the weight of the internals and core in water, preloads and the core drag steady state forces. These calculated static displacements are used as the initial conditions. Without these, some of the masses would be subjected to large accelerations because of the resulting force unbalance.

5. Results and Discussion

The results of analysis consist of shear, moment and axial force of each component which will be used for design loads, and motions for core shroud, fuel alignment plate and core support plate which

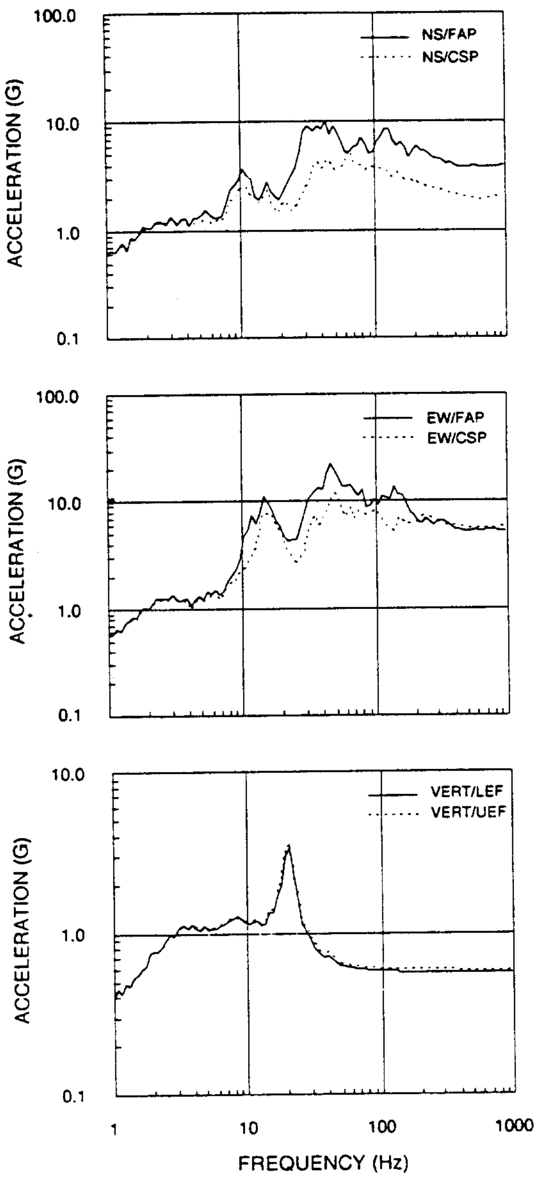


Fig. 4 Response spectra of core plates for 0.3g ground motion.

will be used for the detailed core analysis. Also, the response spectra at several locations of the reactor vessel internals are generated for the ensuing stress analysis to verify the structural integrity of the core support structures.

The response loads of core support structures are summarized in Table 1. It is found that upper flanges of CSB and UGS are most severe response region in all three directions, which is anticipated by the fact that they are.so near to the RV ledge region where the earthquake motion is directly applied. The most response ratio obtained is 46.4 % in UGS upper flange for shear, 60.8 % in CSB upper flange for moment and 36.7 % in CSB lower flange for axial force. The response ratio ranges in 18.8 % to 60.8 % for north-south direction and 15.6 % to 46.1 % for east-west direction. For the vertical direction, the ratio is in the range of 34.1 % to 36.7 %, which corresponds to the ratio of ZPA value (35.1 %) of RV flange motion. This is possible because major frequencies in the vertical direction are high and the ZPA is obtained around those frequencies.

For the subsequent detailed core analysis the response spectra for the fuel alignment plate and core support plate are investigated. An amplification factor of ZPA values from RV flange to FAP is in the range of 3.4 to 5.3. But the spectra values between 1 Hz and 10 Hz are not much amplified and therefore fuel assembly is not anticipated to produce big response because the major fuel assembly modes fall between 1 Hz and 10 Hz.

The result of the core analysis consists of peak spacer grid impact loads, fuel assembly moments, shears and deflected shapes. The impact loads are used to evaluate the structural integrity of spacer grids. The deflected shapes which correspond to peak loading conditions - peak displacement, peak shear and peak moment - are used to calculate stresses using a detailed static model of the fuel assembly. The deflected shapes indicated that the fuel assemblies respond to the seismic excitation by moving back and forth across the core at approximately their first mode natural frequencies.

The spacer grid impact loads and the fuel assembly responses are shown in Table 2. The square root of the sum of the squares of spacer grid impacts are 5675 lbs and 3449 lbs for one-sided and through-grid impacts, respectively. This exceeds the allowables by 28.6 % and 1.6 % for one-sided and through-grid impacts, respectively, which indicates that the fuel assembly design need to be modified for the seismic level increase of 0.3g ground motion. For the axial response of the fuel assembly, the axial force of fuel rods is 758.8 lbs. For both horizontal and vertical directions the response ratio is almost the same as

Table 1 Load summary of reactor vessel internals components

Component	OBE				
	N-S		E-W		Vertical
	Shear	Moment	Shear	Moment	Axial force
CSB Upper Flange	.1845E6	.3311E8	.3055E6	.3134E8	93000
CSB Lower Flange	.5745E5	.4329E7	.1095E6	.7856E7	73000
LSS	.4919E5	.4311E7	.9401E5	.7788E7	58000
UGS Upper Flange	.1720E6	.1422E8	.3017E6	.2520E8	20000
UGS Lower Flange	.5672E5	.2261E7	.7669E5	.2373E7	17000
Tube Sheet Assembly	.3341E5	.2162E7	.3788E5	.2462E7	14000

Component	SSE				
	N-S		E-W		Vertical
	Shear	Moment	Shear	Moment	Axial force
CSB Upper Flange	.4063E6	.4332E8	.7000E6	.5049E8	171000
CSB Lower Flange	.2023E6	.1041E8	.2684E6	.1851E8	133000
LSS	.1477E6	.1027E8	.2098E6	.1830E8	105000
UGS Upper Flange	.2384E6	.1872E8	.5767E6	.4204E8	39000
UGS Lower Flange	.1016E6	.3799E7	.1429E6	.4737E7	32000
Tube Sheet Assembly	.6032E5	.4028E7	.8211E5	.5091E7	28000

Component	SSE(0.3g)				
	N-S		E-W		Vertical
	Shear	Moment	Shear	Moment	Axial force
CSB Lower Flange	.3053E6	.1530E8	.6999E6	.4117E8	199000
LSS	.2310E6	.1514E8	.4860E6	.4097E8	158000
UGS Upper Flange	.3707E6	.3252E8	.6967E6	.6214E8	58000
UGS Lower Flange	.1594E6	.5368E7	.2080E6	.8931E7	48000
Tube Sheet Assembly	.7327E5	.5345E7	.1896E6	.1030E8	41000

Unit : Shear(lbs), Moment(in-lbs), Axial force(lbs)

Table 2 Load summary of fuel assembly

Component	OBE			SSE			SSE(0.3g)		
	N-S	E-W	Vert.	N-S	E-W	Vert.	N-S	E-W	Vert.
Spacer grid									
One-sided impact(lbs)	1455	1275		2606	2801		3696	4307	
Through-grid impact(lbs)	948	894		2091	1771		2214	2644	
Fuel assembly									
Deflection(inch)	1.210	1.160		1.505	1.596		1.706	1.993	
Shear(lbs)	164	199		298	394		417	483	
Moment(lb-inch)	3566	3923		5934	7560		8378	9715	
Axial force(lbs)									
@ Fuel rods			278.5			506.2			758.8
@ UEF			1.1			2.3			3.4
@ LEF			306.8			556.5			835.0
@ Guide tubes			24.9			45.2			67.8

the ratio of input motions, which means that the non-linearity of the fuel assembly response is not significant.

6. Conclusions

Dynamic analyses of the reactor vessel internals and fuel assemblies for the Korean standard nuclear power plant are performed for the 0.3g ground motions of the seismic excitations. The response relations between OBE and SSE and their response characteristics are investigated. By comparing the amplification factors it is found that the reactor vessel internals is more amplified than reactor coolant system components which have more supporting system to absorb the earthquake motion. In this respect, a novel idea for supporting system in the internals components is necessary, which is assumed to be almost impossible due to the function of providing a flow path. Also the results showed that the present design of the fuel assembly does not satisfy the allowables and needs to be modified for the increased seismic level. More detailed evaluations are necessary to define the range of modification from the current design for the 0.3g ground motions.

References

(1) Jhung, M.J., Hwang, J.K. and Kim, Y.S., 1995, "Fuel Assembly Modelling for Dynamic Analysis of Reactor Internals and Core," Journal of the Korean Nuclear Society, Vol.27, No.5, pp.743～752.

(2) Jhung, M.J., 1997, "Axial Response of PWR Fuel Assemblies for Earthquake and Pipe Break Excitations," Structural Engineering and Mechanics, Vol.5, No.2 (to appear).

(3) Blevins, R.D., 1990, Flow-Induced Vibration, 2nd ed., Van Nostrand Reinhold, New York.

(4) Fritz, R.J., 1972, "The Effect of Liquids on the Dynamic Motions of Immersed Solids," Journal of Engineering for Industry, Vol.94, pp.167～173.

(5) Jhung, M.J., 1996, "Hydrodynamic Effect on Dynamic Response of Reactor Vessel Internals," The International Journal of Pressure Vessels and Piping, Vol.69, No.1, pp. 65～74.

(6) Stokes, F.E. and King, R.A., 1978, "PWR Fuel Assembly Dynamic Characteristics," Proceedings of the International Conference on Vibration in Nuclear Plant, Keswick, British Nuclear Energy Society, UK.

(7) Jhung, M.J. and Hwang, W.G., 1994, "Seismic Behavior of Fuel Assembly for Pressurized Water Reactor," Structural Engineering and Mechanics, Vol.2, pp. 157～171.

(8) Gabrielson, V.K., 1966, "SHOCK-A Computer Code for Solving Lumped-Mass Dynamic Systems," Technical Report SCL-DR-65-34, Sandia Laboratories, Livermore, CA, January.