Journal of the Korean Nuclear Society
Volume 30, Number 2, pp.164~172, April 1998

〈Technical Note〉

Modelling of RV Ledge Region for Dynamic Analysis of Coupled Reactor Vessel Internals and Core

Myung J. Jhung
Korea Institute of Nuclear Safety
19 Kusong-dong, Yusong-gu, Taejon 305-338, Korea

(Received March 26, 1996)

Abstract

This paper presents the detailed modelling of reactor vessel ledge region for the dynamic analysis of the coupled internals and core model. The dynamic responses due to earthquake and pipe break are calculated using the input motions of reactor vessel taken from Ulchin nuclear power plant units 3 and 4. Two different representations for detailed and simplified models of the RV ledge region are made. The dynamic responses of the reactor internals components are compared between them. Response characteristics are reported and simplified model is suggested for earthquake and pipe break analysis for the future design of the reactor internals.

1. Introduction

The core support barrel (CSB) assembly, the major structural member of the reactor internals, is supported at its upper end by the upper flange, which rests on a ledge in the reactor vessel (RV). The reactor vessel ledge, closure head, upper guide structure (UGS) barrel flange and holddown ring are slotted in locations corresponding to the alignment key locations to provide alignment between these components in the reactor vessel flange region (Fig. 1).

To model this region for the dynamic response analysis, it was generally modeled as one mass point, which is reported to generate too conservative design loads. Therefore, it is necessary to account for the interactions between the CSB upper flange, UGS upper flange, hold-down ring and the reactor vessel ledge. This may

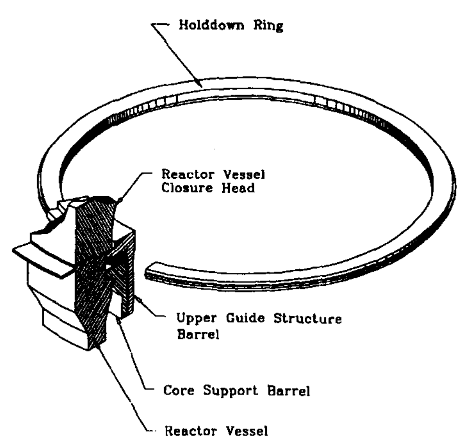

Fig. 1. Schematic Diagram of Reactor Vessel Ledge Region

be done using the nonlinear, hysteresis and friction elements.

In this paper, two different representations of the RV ledge region such as detailed and simplified models are made. The dynamic responses of the reactor internals components are compared between them. Response characteristics are reported and simplified model is suggested for earthquake and pipe break analysis for the future design of the reactor internals.

2. Model Development

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 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. However, for every analys is performed, this assumption should be verified by confirming that the relative deflections of component are in fact smaller than existing gaps.

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 internal components. For the beam element connecting two nodes, properties are calculated for moments of inertia, cross-sectional areas, effective shear areas, stiffnesses and length.

Stiffnesses for the complex internal structures such as UGS and CSB flanges, CSB snubber, hold-down ring and control element assembly

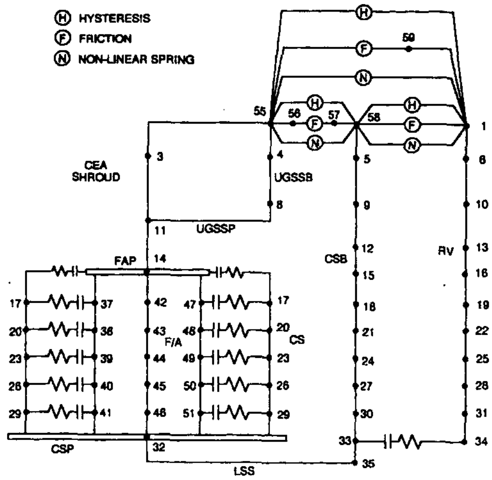

Fig. 2. Lumped Mass Model of Reactor Vessel Internals and Core : Detailed Representation of the Ledge Region

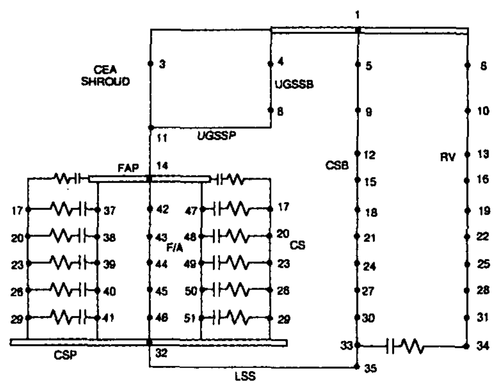

Fig. 3. Lumped Mass Model of Reactor Vessel Internals and Core : Simplified Representation of the RV Ledge Region

(CEA) 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 structures. A dynamically equivalent representation of the CEA shroud is included in the model. This representation is based on a frequency analysis of

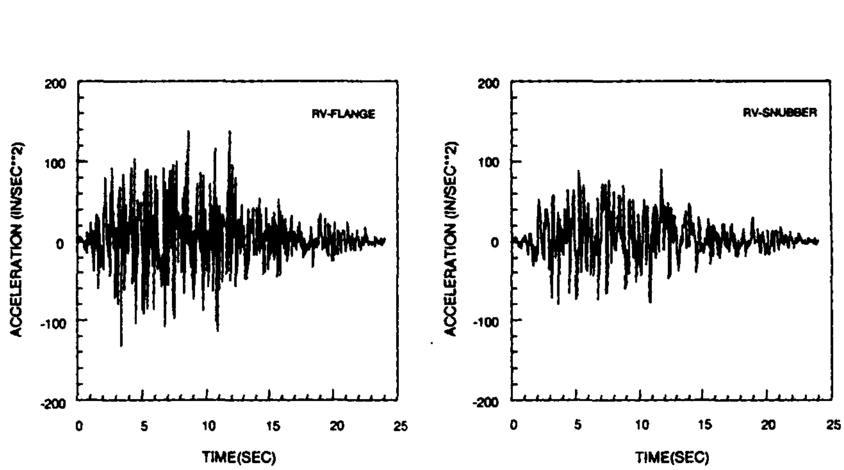

Fig. 4. Acceleration Time Histories of Reactor Vessel for SSE(0.2g)

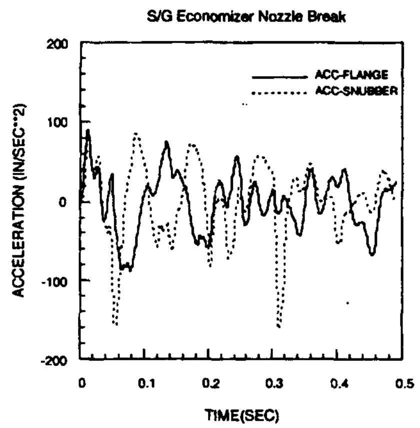

Fig. 5. Acceleration Time Histories of Reactor Vessel for Pipe Break

the detailed finite element model [1, 2]. The CSB upper region is modeled to account for the possible interactions between the CSB upper flange, UGS upper flange, hold-down ring and the reactor vessel ledge using the nonlinear, hysteresis and friction elements. Also, to see the effect of the representing this region as one mass point, a simplified model is made.

A typical coupled internals and core models are shown in Figs. 2 and 3. The actual arrangement and detail in the model may vary with the function of plant design, and the magnitude and nature of the excitation.

3. Analysis

The forcing function to the 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 acceleration time histories of RV flange and snubber which were generated from the reactor coolant system analysis are shown in Figs. 4 and 5.

The maximum free field horizontal ground accelerations at the foundation level of 0.2g for the safe shutdown earthquake (SSE) are used to get the reactor vessel motions and the input excitations of the SSE are increased by 50 % to get the response characteristics for the possible operating basis earthquake elimination, which is

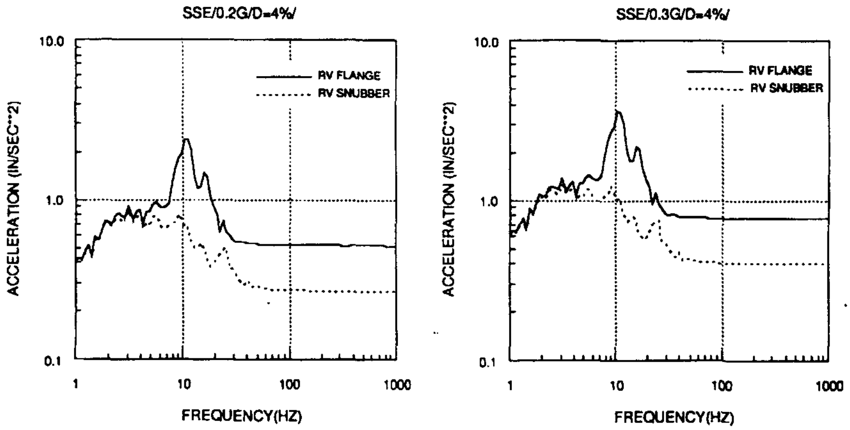

Fig. 6. Response Spectra of Reactor Vessel for SSE

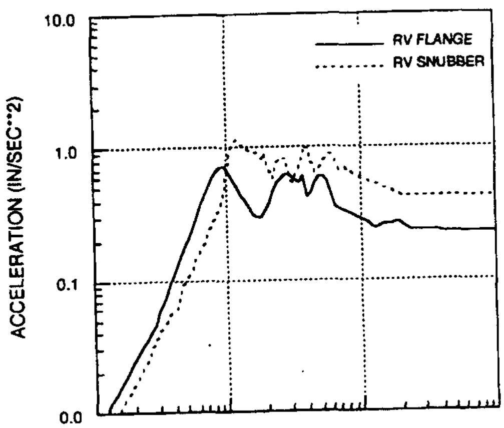

Fig. 7. Response Spectra of Reactor Vessel for Pipe Break

being studied for the future plant design [3].

In the recent design of nuclear power plants, main coolant loop double ended guillotine breaks could be eliminated from the design basis by introducing leak-before-break (LBB) concept. Instead branch line pipe breaks are considered as one of the Level D service loadings. It is anticipated that all pipe breaks with a diameter of 10 inches or over be not considered as design basis any more. But the pipe breaks of 12 SCH180 economizer feedwater line in the secondary side were reported in many plants because of water hammer. Therefore this break should be design basis even though elimination of all other high energy piping systems with a diameter of 10 inches or over is accepted based upon current LBB evaluations [4, 5].

The acceleration time histories of RV flange and snubber for SSE and economizer feedwater line break and its corresponding spectra are shown in Figs. 4 through 7. The maximum accelerations for the SSE are 197.0 in/sec^2 (0.510g) at 8.485 seconds and 102.6 in/sec^2 (0.266g) at 7.088 seconds for the RV flange and snubber elevations, respectively. For the pipe break, the maximum values are 88.7 in/sec^2 (0.230g) at 0.012 seconds and 161.5 in/sec^2 (0.418g) at 0.311 seconds for the RV flange and snubber elevations, respectively. A different characteristics of the input motions are noticed at the RV region : for seismic case the RV flange motion is more severe than RV snubber, but the opposite is true for the pipe break case.

The response of the internals is computed by the SHOCK code [6], 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

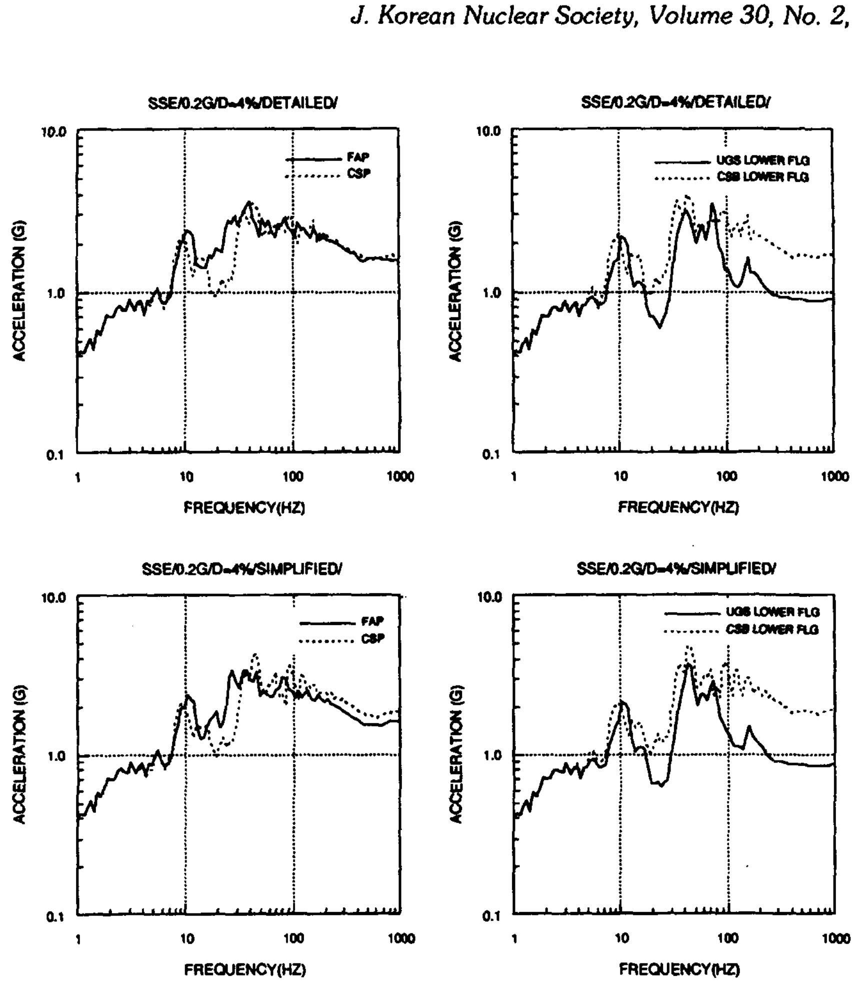

Fig. 8. Response Spectra of Core Plates and Flanges for SSE(0.2g)

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 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.

4. Results and Discussion

The results of analysis consist of minimum and maximum values of shears and moments of each component which will be used for design loads, and motions for fuel alignment plate and core support plate which will be used for the detailed core analysis. Also, the response spectra at several locations of the reactor internals are generated for the ensuing stress analysis for the components.

The design loads of each component are summarized in Table 1. The seismic loads for the detailed and simplified models are almost same except for the CSB upper flange, where the load for detailed representation is smaller than that of

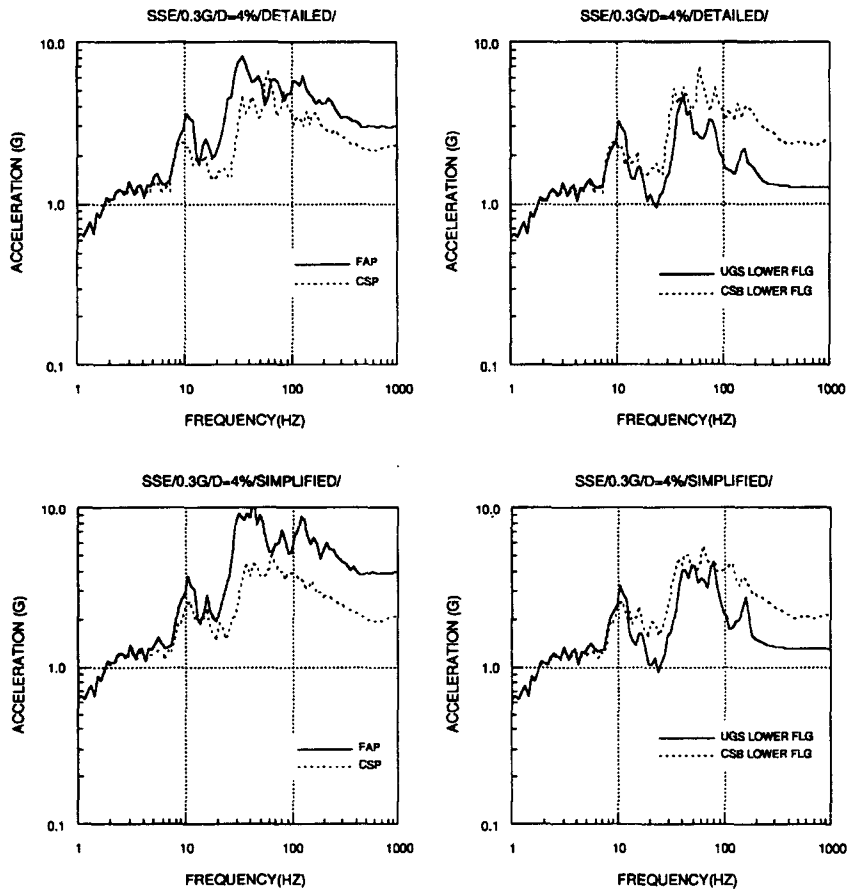

Fig. 9. Response Spectra of Core Plates and Flanges for SSE(0.3g)

simplified model. For the pipe break responses, the loads from detailed model is smaller than those of simplified model. This indicates that the simplified representation of the reactor vessel ledge region is suggested for the dynamic analysis due to pipe break.

For the subsequent detailed core analysis the response spectra for the fuel alignment plate and core support plate are compared. The spectra values between detailed and simplified models are almost same for the regions between 1 Hz and 10 Hz (Figs. 8 through 10), which contribute most of the major fuel assembly modes [7]. From this comparison, it is assumed that the different modelling representations of the RV ledge region do not affect the detailed core analysis even though there is a little difference in the spectra values for the higher frequency region.

To verify the structural integrity of the core support structure, the response spectrum analysis using the spectra generated from the coupled internals and core model is performed. The response spectra are compared and the spectra from simplified model is generally higher than those from detailed model for the higher frequency region, or over 40 Hz (Figs. 8 through 10). Since most of the major modes for internals components are below 40 Hz, it is therefore not anticipated

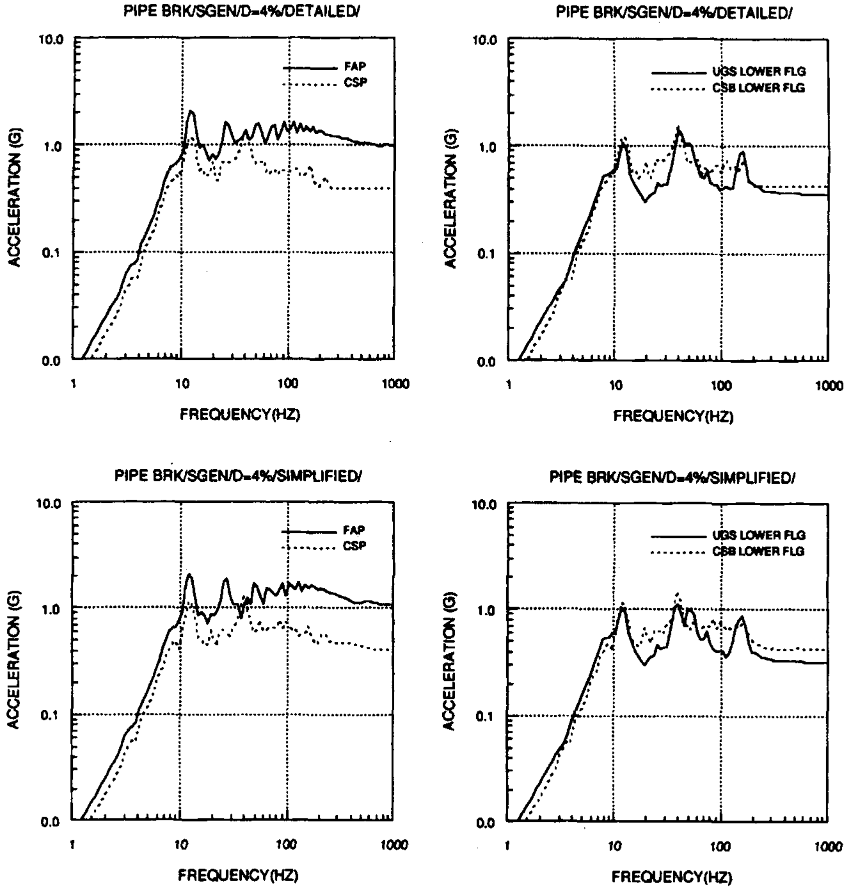

Fig. 10. Response Spectra of Core Plates and Flanges for Pipe Break

that the detailed model reduce the stress intensity for the SSE significantly.

5. Conclusions

The modelling method of the RV ledge is studied for the coupled reactor vessel internals and core analysis due to earthquake and pipe break excitations. The following conclusions were reached:

1. The simplified representations of the reactor vessel ledge region are suggested for the dynamic analysis due to pipe break and earthquake.
2. The modelling representation of the RV ledge region do not have any effect on the detailed core analysis
3. By comparison of spectra values for the ensuing response spectrum analysis, it is anticipated the detailed model doesn't reduce the stress intensity for the SSE.

Table 1. Response Comparisons Between Detailed and Simplified Models

COMPONENT	SHEAR FORCE (LBS)						MOMENT (LBS-IN)					
	SSE-0.2G		SSE-0.3G		SGEN		SSE-0.2G		SSE-0.3G		SGEN	
	SIMP.	DET.	SIMP.	DET.	SIMP.	DET.	SIMP.	DET.	SIMP.	DET.	SIMP.	DET.
CSB UPPER FLANGE	.4063E6	.3798E6	.5514E6	.4649E6	.3306E6	.3059E6	.4332E8	.4067E8	.5450E8	.4471E8	.2214E8	.2115E8
CSB LOWER FLANGE	.2033E6	.1927E6	.3053E6	.3249E6	.9027E5	.8838E5	.1041E8	.1113E8	.1530E8	.1510E8	.6728E7	.6246E7
UGS UPPER FLANGE	.2384E6	.2525E6	.3707E6	.3579E6	.2863E6	.2779E6	.1872E8	.1926E8	.3252E8	.2874E8	.2052E8	.2184E8
UGS LOWER FLANGE	.1016E6	.1089E6	.1594E6	.1550E6	.6098E5	.6056E5	.3799E7	.3569E7	.5368E7	.5418E7	.2015E7	.1991E7
LSS	.1477E6	.1400E6	.2310E6	.2440E6	.7207E5	.7041E5	.1027E8	.1106E8	.1514E8	.1477E8	.6782E7	.6236E7
CORE SHROUD	.8051E5	.8287E5	.1336E6	.1275E6	.5497E5	.5256E5	.8696E7	.9189E7	.1300E8	.1337E8	.5877E7	.5583E7
CEA SHROUD	.5159E5	.5806E5	.8004E5	.8077E5	.5279E5	.5118E5	.1667E7	.1871E7	.2407E7	.2466E7	.1394E7	.1331E7
TUBE BANK	.6032E5	.6241E5	.7327E5	.7520E5	.2670E5	.2883E5	.4028E7	.4186E7	.5345E7	.5386E7	.1500E7	.1625E7
F/A-CORE SHROUD	.5720E5	.6280E5	.9303E5	.9472E5	0.0	0.0	-	-	-	-	-	-

References

1. Jhung, M.J., Choi, S. and Song, H.G., "Analytical Study on the Vibration Characteristics of Control Element Assembly Shroud," *Proceedings of the International Conference on Structural Dynamic Modelling*, pp.69-78, Milton Keynes, UK, July (1993).
2. Jhung, M.J. and Choi, S., "Experimental Study on the Vibration Characteristics of Control Element Assembly Shroud," *Proceedings of the International Conference on Vibration Engineering*, pp.465-470, Beijing, China, June (1994).
3. USNRC, Policy Issue SECY-93-087, US Nuclear Regulatory Commission, April 2, (1993).
4. USNRC, "Evaluation of Potential for Pipe Breaks," NUREG-1061, Vol.3, US Nuclear Regulatory Commission, November (1984).
5. Roos, E., et al., "Assessment of Large Scale Pipe Tests by Fracture Mechanics Approximation Procedures with Regard to Leak-Before-Break," *Nuclear Engineering and Design*, Vol.112, pp.183-195, (1989).
6. Gabrielson, V.K., SHOCK-A *Computer Code for Solving Lumped-Mass Dynamic Systems*, Report SCL-DR-65-34, Sandia Laboratories, Albuquerque, NM, (1966).
7. Jhung, M.J. and Hwang, W.G., "Seismic Behavior of Fuel Assembly for Pressurized Water Reactor," *Structural Engineering and Mechanics*, Vol.2, No.2, pp.157-171, (1994).