Journal of the Korean Nuclear Society
Volume 27, Number 5, October 1995

Fuel Assembly Modelling for Dynamic Analysis of Reactor Internals and Core

Myung Jo Jhung, Jong Keun Hwang, and Yeon Seung Kim
Korea Atomic Energy Research Institute
(Received March 27, 1995)

원자로 내부구조물과 노심의 동적해석을 위한 핵연료집합체의 모델링

정명조 · 황종근 · 김연승
한국원자력연구소
(1995. 3. 27 접수)

Abstract

This paper investigates the effects of fuel groupings in the coupled internals and core model on the internals and fuel responses due to pipe breaks. The 177 fuel assemblies for Korean standard nuclear power plant are grouped into several stick models and the responses of internals components are calculated. The analysis results show that the fuel model groupings in the coupled internals and core model have no significant effects on the internals and fuel responses for pipe break excitation. Also, in order to determine the feasibility of constructing a single equivalent stick representation of two or more adjacent fuel bundles, the reduced models, each of which employs a different stiffness lumping rule, are constructed. It is shown that the equivalent stiffness calculated to get the first natural frequency of the original model while preserving net gap between grouping centers gives the minimum modelling error.

요 약

본 논문은 배관파단에 대한 원자로 내부구조물의 해석시 사용되는 원자로 내부구조물과 노심의 커플(couple)된 모델에서 핵연료집합체의 grouping수에 따른 동적 응답의 영향을 조사한 것이다. 177개의 핵연료집합체를 1, 3, 5 그리고 7개의 그룹으로 나누어 모델링하였고 그 각각에 대한 응답을 구하였다. 해석결과 원자로 내부구조물과 핵연료집합체의 배관파단에 대한 응답은 핵연료집합체의 grouping수에 거의 영향을 받지 않음을 알 수 있었다. 또한 핵연료집합체의 해석시 사용되는 상세모델에서 2개이상의 이웃하는 핵연료다발을 하나의 등가모델로 나타내는 방법을 연구한 결과 집합체의 1차모드 주파수와 일치하는 등가스프링을 사용하고 각 핵연료다발사이의 간격을 그대로 유지했을 때의 모델이 원래의 응답과 가장 잘 일치함을 보였다.

1. Introduction

For the dynamic response of reactor internals under pipe break or seismic excitation, the coupled model of internals and core is used, which uses the reduced core model by lumping all fuel assemblies into several groups. The fuel assembly responses are obtained from the detailed core model analysis using the forcing function generated from the coupled internals and core analysis.

This paper investigates the modification to the fuel model groupings in the coupled internals and core model. It determines the effects of alternative fuel grouping in the internals model might have on core model impacting loads. Fuel is grouped by lumping the peripheral bundles into sticks adjacent to the core shroud. In several analyses, the next-row-in was also grouped and allowed to impact the peripheral bundles. The remaining fuel was lumped at the middle and assumed not to impact. This precluded the transmission of impacting across the core and sometimes resulted in large fuel displacements because of lack of support. For this study, it is decided to model impacting across the core. This is done by breaking the core into groups, representing each with a stick in the internals model and allowing impacting to occur between adjacent groups and with the core shroud. The 177 fuel assemblies for Korean standard nuclear power plant [1] are grouped into several stick models and the responses of internals components are calculated.

To get the dynamic response of fuel assembly, it is necessary to model all fuel assemblies individually. In this case, the number of degrees of freedom may be a problem due to computing time for time history analysis. This necessitates the use of a reduced model for detailed core analysis. In order to determine the feasibility of constructing a single equivalent stick representation of two or more adjacent fuel bundles for the detailed core analysis, an eleven bundle single load path model is assembled to determine the behavior of the physical system. Four reduced models are employed which consist of five sticks to represent the eleven physical bundles of the actual system. In the four reduced models, the three interior sticks lump three physical sticks each. The remaining two physically peripheral sticks of the reduced models represent the two physically peripheral sticks of the actual physical system.

Each of four reduced models employs a different equivalent stiffness and is subjected to the forcing inputs as the actual model to determine which stiffness lumping rule provide the best synthesis of the true core shroud loads.

2. Model Development

2.1. Coupled Internals and Core Model

Core model in the coupled internals and core 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 fuel assembly groupings and choosing stiffness values to adequately characterize its beam response and contacting under dynamic loading. Fuel assemblies are combined into various groupings. There are various core region representations and Figure 1 shows a core region with various 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 [2].

The stiffness properties of each fuel grouping are determined by combining the individual fuel assembly stiffnesses added in parallel. Individual fuel assemblies are 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 same as those used in the detailed core model.

The fuel assembly properties are developed according to the number of fuel groupings using the individual fuel properties multiplied by the number of fuel assemblies. The models for one, three, five and seven fuel groupings are shown in Figures 1 and 2. The model properties are calculated according to the number of fuel assemblies included in the groupings.

2.2. Detailed Core Model

To get the detailed response of fuel assembly, it is necessary to perform a detailed core analysis which uses core plates and core shroud motions from the coupled internals and core analysis. In the detailed core model, each fuel assembly is modeled individually 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 pos-

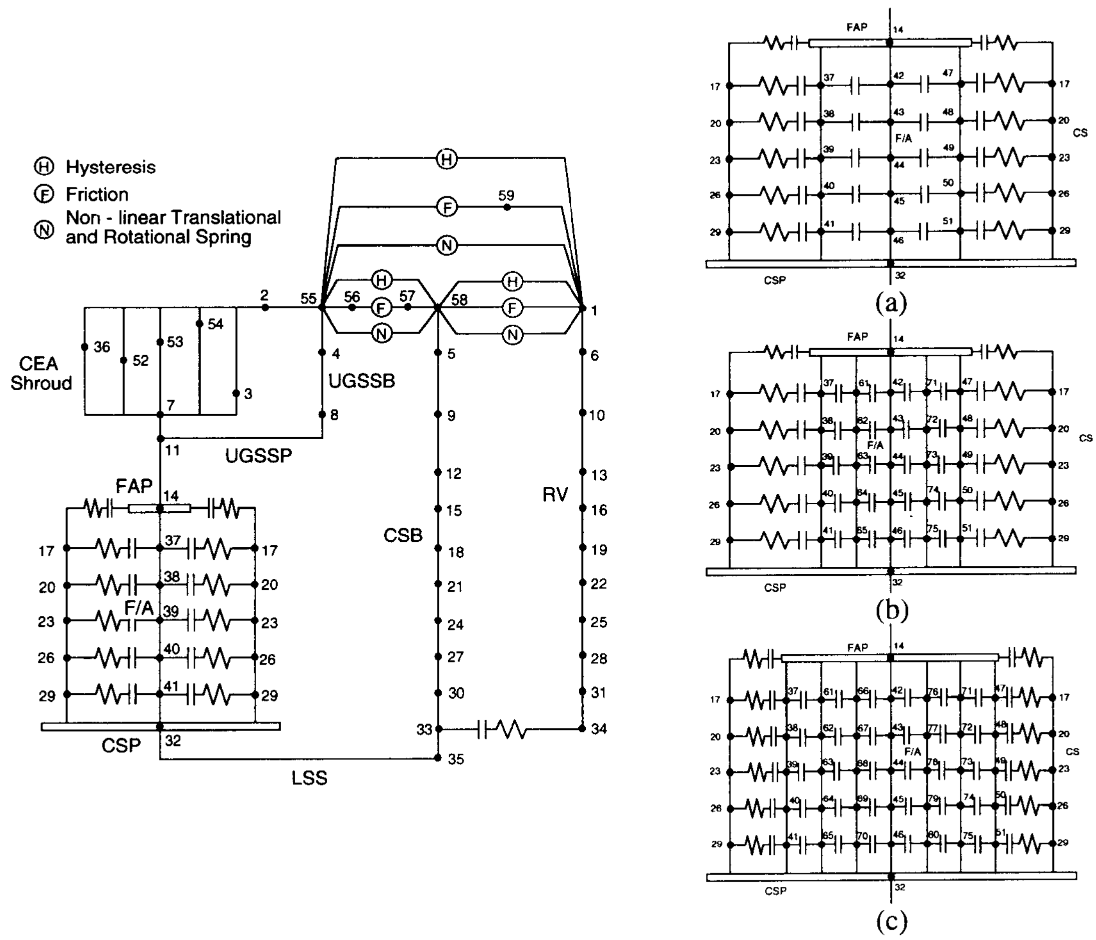

Fig. 1. Coupled Internals and Core Model with Various Fuel Assembly Groupings (a) 3 Groupings, (b) 5 Groupings, (c) 7 Groupings

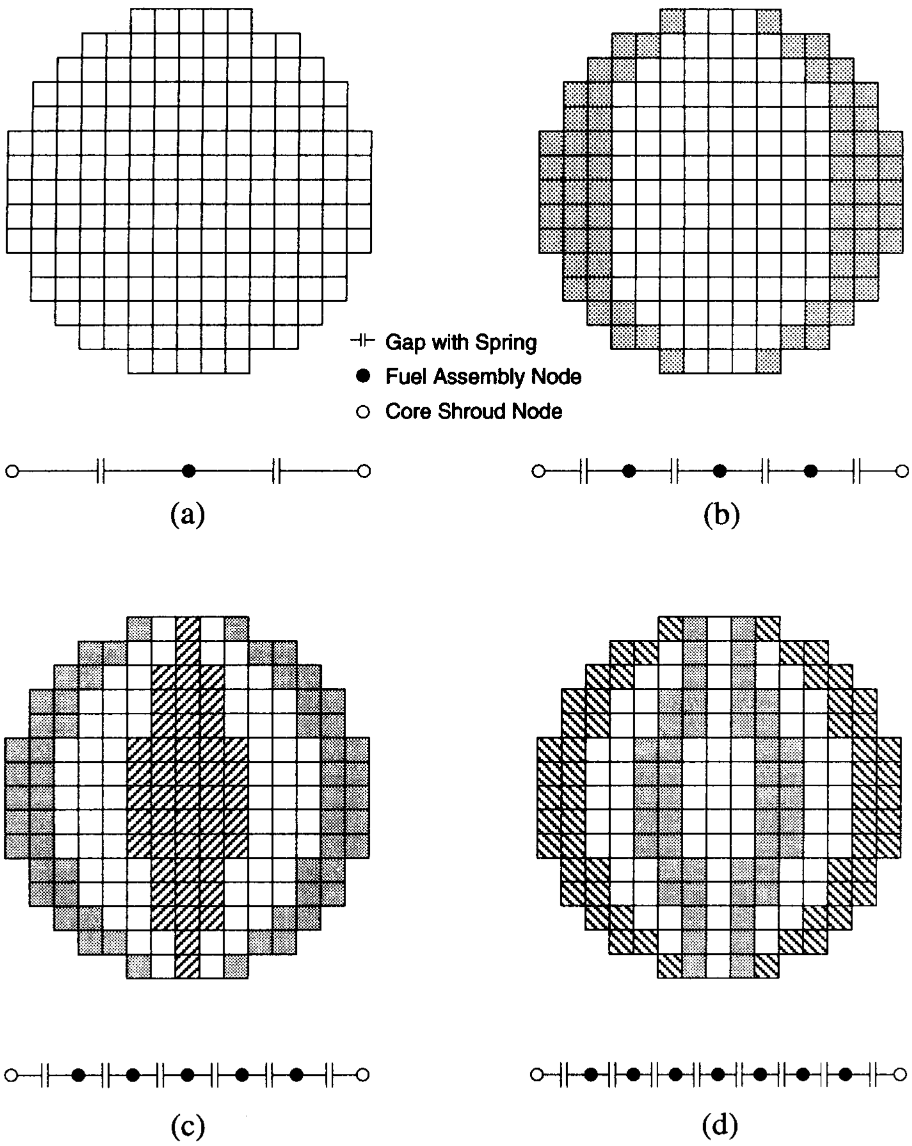

Fig. 2. Various Fuel Assembly Groupings (a) 1 Grouping (b) 3 Groupings (c) 5 Groupings (d) 7 Groupings

sible spacer grid impacting. Nonlinear spring couplings are used to simulate the gaps in the core. Each spacer grid is characterized by the single load path model which represents the load paths associated with one-sided impact [3].

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 static 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.

Hydrodynamic (diagonal coupling coefficients) mass was added to the structural mass to obtain the proper natural frequency in water. 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 [4], 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 pluck vibration, pluck impact, spacer grid compression, and spacer grid section drop tests provide data used in determining the spacer grid impacting parameters. The eleven bundle single load path model and reduced model are shown in Figures 3 and 4. To represent the elastic and inertial properties of three adjacent fuel bundles in one equivalent stick, the sticks of reduced models are assigned three times the flexural rigidity (EI), mass and support rotational stiffness of a single bundle. Four different rules to calculate the equivalent one sided spacer grid stiffness and gap are used.

MODEL 1

Model 1 preserves the series addition of the stiffness of all eleven adjacent fuel bundles with that of the five sticks of the reduced model. Also, it preserves the net gap between grouping centers. It uses the same equivalent spring in four places.

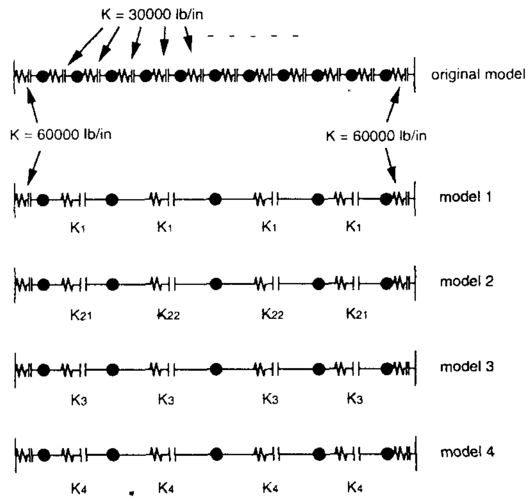

Fig. 3. Equivalent Stick Model Representations of Fuel Bundles

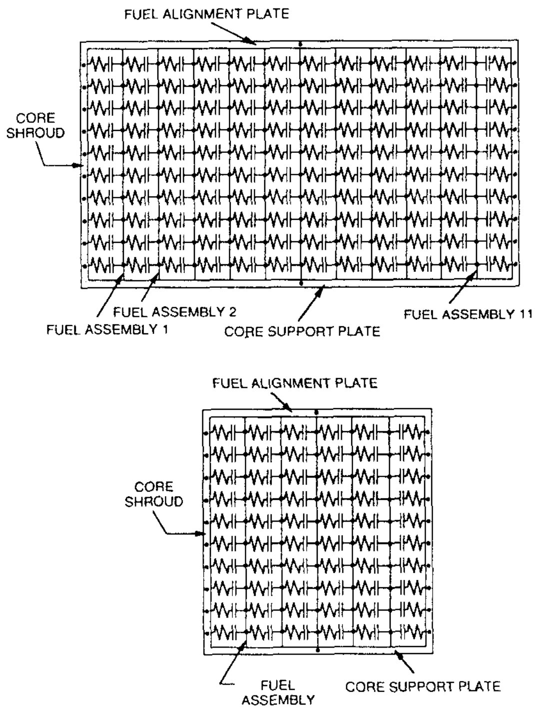

Fig. 4. Detailed Core Models of 11 Sticks for 11 Fuel Bundles (upper) and its Corresponding Reduced 5 Sticks (lower)

$$\frac{1}{K_{TOTAL}}=\frac{2}{60000}+\frac{10}{30000}$$

$\therefore$ K_{TOTAL} = 2727 lb/in = K_{1TOTAL}

$$\frac{1}{K_{1TOTAL}}=\frac{2}{60000}+\frac{4}{K_1}=\frac{1}{2727}$$

$\therefore$ $K_1=11999$ lb/in

MODEL 2

Model 2 preserves the total series addition of stiffness of the physical model with that of the reduced model. Also, it preserves the net gap between grouping centers.

$$\frac{1}{K_{21}}=\frac{1}{30000}+\frac{1}{30000}$$

$\therefore$ $K_{21}=15000$ lb/in

$$\frac{1}{K_{22}}=\frac{1}{30000}+\frac{1}{30000}+\frac{1}{30000}$$

$\therefore$ $K_{22}=10000$ lb/in

$$\frac{1}{K_{2TOTAL}}=\frac{2}{60000}+\frac{2}{15000}+\frac{2}{10000}$$

$\therefore$ $K_{2TOTAL}=2727$ lb/in

MODEL 3

Model 3 preserves the first mode frequency of the interior nine (out of eleven) physical bundles with that of the three interior sticks of the reduced model Also, it preserves the net gap between grouping centers. It uses the same equivalent spring in four places. The ith natural frequency of N equal masses and (N+1) equal springs is [5]

$$f_i=\frac{a_{N,i}}{2\pi}\sqrt{\frac{k}{m}}$$

where $\alpha_{N,i}=2\sin(\frac{i}{N+1}\frac{\pi}{2})$, k and m are stiffness and mass, respectively. The first mode frequency for original model of 11 sticks is calculated as;

$m=166.73/386.4=0.4315$ lb_m

$k=30000$ lb/in

$N=9$

$$a_{9,1}=2\sin(\frac{1}{9+1}\frac{\pi}{2})=0.312869,$$

$$\therefore\ f_1=\frac{0.312869}{2\pi}\sqrt{\frac{30000}{0.4315}}=13.1296\ \text{Hz}.$$

For the reduced model with 4 equivalent springs

$m=(0.4315)(3)=1.2945$ lb_m

$k=30000$ lb/in

$N=3$

$$a_{3,1}=2\sin(\frac{1}{3+1}\frac{\pi}{2})=0.765367,$$

$$f_1=\frac{0.765367}{2\pi}\sqrt{\frac{K_3}{0.4315}}=13.1296\ \text{Hz}.$$

$\therefore$ $K_3=15039$ lb/in

$$K_{3TOTAL}=(\frac{2}{60000}+\frac{4}{15039})^{-1}=2727\ \text{lb/in}.$$

MODEL 4

Model 4 preserves the physical one sided stiffness and gap as the effective stiffness and gap of the reduced model. It assumes that the relative distance between adjacent bundles in groupings remain fixed as if rigidly connected.

$K_4=30000$ lb/in

$$K_{4TOTAL}=(\frac{2}{60000}+\frac{4}{30000})^{-1}=6000\ \text{lb/in}$$

3. Dynamic Analysis

3.1. Forcing Functions

In the recent design of nuclear power plants, main coolant loop double ended guillotine breaks are eliminated from the design basis because of leak-before-break (LBB) concept. Instead branch line pipe breaks are considered as one of the Level D ser vice loadings. It is anticipated that all pipe breaks with a diameter of 10 inches or over benot 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 due to 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 [6, 7].

The response time histories of RV flange and snubber for economizer feedwater line break are input to the internals analysis. They are obtained from the reactor coolant system analysis. The acceleration time history of RV flange is shown in Figure 5.

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. The response time history of core plates and core shroud are used which calculated in the coupled internals and core analysis with 3 fuel groupings. Because the spacer grid impact loads don't occur for the economizer feedwater line break considered in this analysis, the forcing function is multiplied by 5.0 and input to the detailed core model. The response spectra plots of core plates are shown in Figure 6.

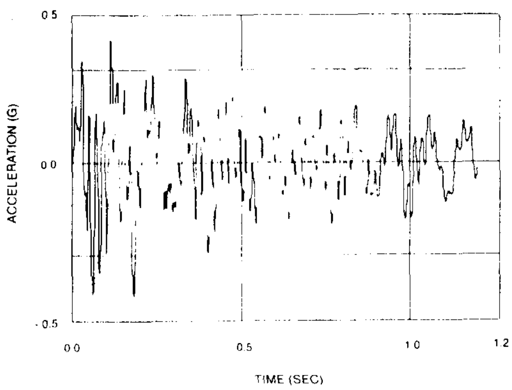

Fig. 5. Acceleration Time History of RV Flange for Pipe Break Excitation

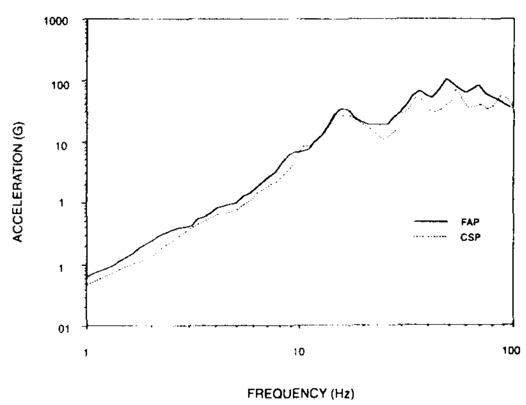

Fig. 6. Response Spectra of Core Plates for Pipe Break Excitation

3.2. Dynamic Responses

The response of the 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 [9]. 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

4.1. Internals Responses

The results of analysis consist of the displacement, velocity and acceleration time histories of fuel alignment plate, core support plate and core shroud nodes which will be used for the detailed core analysis, and minimum and maximum values of shears and moments of each coupling which will be used for design loads. The maximum shear loads and moments for internals components are summarized in Tables 1 and 2. As shown, the number of fuel model grouping in the coupled internals and core model has no significant effect on the response of internals for the pipe break excitation.

4.2. Core Responses

The time history analyses of the eleven stick physical core model and four reduced models are performed. The maximum spacer grid loads are summarized in Table 3. Since it is only possible to compare directly the spacer grid loads in the peripheral bun-

Table 1. Summary of Maximum Shear Loads of Internals (lbs)

Component	Number of Fuel Groupings			
	1	3	5	7
CSB Upper Flange	.7051E6	.7051E6	.7054E6	.7072E6
CSB Upper Cylinder	.7415E6	.7415E6	.7415E6	.7415E6
CSB Nozzle Cylinder	.5061E6	.5061E6	.5061E6	.5061E6
CSB Center Cylinder	.4280E6	.4280E6	.4280E6	.4280E6
CSB Lower Cylinder	.3742E6	.3742E6	.3743E6	.3743E6
CSB Lower Flange	.3931E6	.3931E6	.3932E6	.3932E6
UGS Upper Flange	.9329E6	.9330E6	.9331E6	.9331E6
UGS Lower Flange	.1752E6	.1752E6	.1752E6	.1752E6
LSS	.3250E6	.3250E6	.3250E6	.3250E6
CS	.3488E6	.3488E6	.3488E6	.3488E6
CEA Shroud (at Base)	.6769E5	.6769E5	.6769E5	.6769E5
CSB Snubber	.6227E6	.6227E6	.6227E6	.6227E6

Table 2. Summary of Maximum Moments of Internals (lbs-in)

Component	Number of Fuel Groupings			
	1	3	5	7
CSB Upper Flange	.4418E8	.4418E8	.4418E8	.4418E8
CSB Upper Cylinder	.4632E8	.4632E8	.4633E8	.4633E8
CSB Nozzle Cylinder	.5032E8	.5032E8	.5033E8	.5033E8
CSB Center Cylinder	.4774E8	.4773E8	.4775E8	.4774E8
CSB Lower Cylinder	.3069E8	.3069E8	.3069E8	.3069E8
CSB Lower Flange	.2589E8	.2589E8	.2589E8	.2589E8
UGS Upper Flange	.5660E8	.5660E8	.5660E8	.5660E8
UGS Lower Flange	.8344E7	.8344E7	.8345E7	.8344E7
LSS	.2526E8	.2526E8	.2527E8	.2526E8
CS	.2451E8	.2451E8	.2451E8	.2451E8
CEA Shroud (at Base)	.6155E7	.6155E7	.6155E7	.6155E7

dles striking the core shroud between physical and reduced models, the differences between the actual core shroud loads predicted by the actual model and the loads by the reduced models were computed and their absolute values summed as an indication of total modelling error.

The reduced model resulting in the smallest computed total error was model 3 which was constructed to preserve the first mode frequency of the physical bundles with that of the reduced model and also preserve net gap between grouping centers. It should be noted that the total modelling errors for model 1 and model 2 were also small, whereas the error of model 4 is so big. While reduced models 1, 2 and 3 were judged to be near equally appropriate, model 3 was judged to be the best modelling technique.

The total core shroud load modelling errors versus effective spring stiffness are plotted in Figure 7 and it shows that model 3 provides the most accurate simulation of the actual core shroud loads.

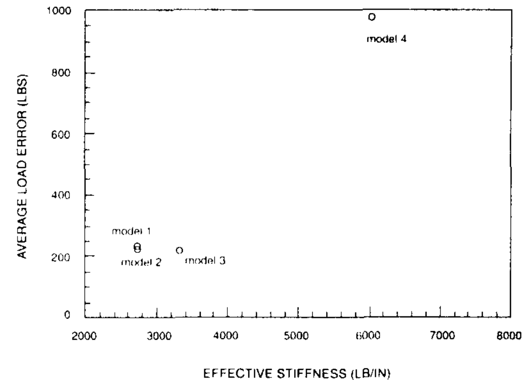

Fig. 7. Total Core Shroud Load Error vs. Effective Stiffness

5. Conclusions

Several coupled internals and core models were employed to compare the response loads of internals and fuel assembly for various fuel groupings. The analysis results show that the fuel model groupings in the coupled internals and core model have no significant effects on the internals responses for pipe break excitation.

To construct a single equivalent stick representation of two or more adjacent fuel bundles, it is determined to be the best method to preserve the first mode frequency of the physical bundles with that of

Table 3. Summary of Spacer Grid Impact Loads (lbs)

NODE NO.	ORIGINAL MODEL		REDUCED MODEL NO. 1		2		3		4		DIFFERENCE |ORIGINAL-MODEL NO.| 1		2		3		4	
	L	R	L	R	L	R	L	R	L	R	L	R	L	R	L	R	L	R
13	0	782	0	773	0	775	0	775	1221	862	0	9	0	7	0	7	1221	80
12	1986	632	2416	571	2173	573	2139	573	3541	640	430	61	187	59	153	59	1555	8
11	3173	479	2527	580	2369	578	2369	577	4161	580	646	101	804	99	804	98	988	101
10	529	642	1917	756	1904	765	1860	761	3779	932	138	114	1375	123	1331	119	3250	290
9	0	570	0	692	0	708	0	703	2228	820	0	122	0	138	0	133	2228	250
8	0	323	0	230	0	230	0	238	2691	479	0	93	0	93	0	85	2691	156
7	236	423	0	372	0	373	0	373	3569	487	230	51	236	50	236	50	3333	64
6	1703	404	1912	345	1915	343	1915	342	3137	346	209	59	212	61	212	62	1434	58
5	1557	562	2474	545	2368	535	2332	535	2626	548	917	17	811	27	775	27	1069	14
4	1912	932	1793	990	1781	991	1781	991	2680	932	119	58	131	59	131	59	768	0
TOTAL ERROR											3945	685	3756	716	3642	699	18537	1021
TOTAL ERROR(L+R)												4630		4472		4341		19558
AVERAGE ERROR												232		224		217		978

L : left side impact, R : right side impact.

the reduced model and also preserve net gap between grouping centers.

References

1. KEPCO, K-SSAR : Korean Standard Nuclear Power Plant Safety Analysis Report, Korea Electric Power Corporation, Seoul (1991)
2. Jhung, M.J., Park, K.B., Hwang, W.G., "Dynamic Response of Reactor Internals to Pipe Breaks," *Nuclear Engineering and Design*, Vol. 152, pp. 79−90 (1994)
3. Jhung, M.J., Hwang, W.G., "Seismic Behavior of Fuel Assembly for Pressurized Water Reactor," *Structural Engineering and Mechanics*, Vol. 2, No. 2, pp. 157−171 (1994)
4. Stokes, F.E. and King, R.A., "PWR Fuel Assembly Dynamic Characteristics," BNES, Vibration in Nuclear Plants, Keswick, UK (1978)
5. Blevins, R.D., *Formulas for Natural Frequency* and Mode Shapes, Van Nostrand Reinhold Company (1979)
6. USNRC, "Evaluation of Potential for Pipe Breaks," NUREG-1061, Vol. 3, US Nuclear Regulatory Commission, November (1984)
7. 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)
8. Gabrielson, V.K., "SHOCK-A Computer Code for Solving Lumped-Mass Dynamic Systems," Technical Report SCL-DR-65-34, Sandia Laboratories, Livermore, CA, January (1966)
9. Kuo, S.S., *Computer Applications of Numerical Methods*, Addison-Wesley, New York (1972)